RINGHALS KRAFTSTATION
AGGREGAT 3

Koncessionsansökan
enligt atomenergilagen
med
preliminär säkerhetsrapport

BAND 1

Statens Vattenfallsverk

RINGHALS KRAFTSTATION
AGGREGAT 3

Koncessionsansökan
enligt atomenergilagen
med
preliminär säkerhetsrapport

BAND 1
Kapitel 1 och 2

Statens Vattenfallsverk

Nr

Avskrift

Till K O N U N G E N

Under åberopande av lagen den 1 juni 1956 om rätt att utvinna atomenergi m m får Statens Vattenfallsverk härmed hos Eders Kungl. Maj:t i underdånighet anhålla om tillstånd att på sätt, som av

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Vattenfall,
angående ansökan om tillstånd till uppförande och drift av atomreaktor m m för det tredje aggregatet vid Ringhals kraftstation.

310 4.57 30 000

bifogade handlingar närmare framgår, dels uppföra, innehava och driva en atomreaktor, benämnd Ringhals 3, dels för atomreaktorns drift förvärva, innehava och använda erforderlig mängd uran. Reaktorn är avsedd att i ett tredje aggregat till Ringhals kraftstation utgöra energikälla för alstrande av elektrisk kraft. Avsikten är att aggregatet skall ingå som produktionskälla i den svenska kraftproduktionsapparaten.

Efter underhandskontakter med delegationens för atomenergifrågor reaktorförläggningskommitté har Vattenfallsverket meddelats tillstånd att utföra vissa avsnitt av den tekniska beskrivningen på engelska.

Vattenfallsverket får härjämte anhålla, att Eder Kungl. Maj:t måtte förklara verket berättigat att, i den ordning lagen den 12 maj 1917 om expropriation föreskriver, genom expropriation med äganderätt förvärva fastigheten Biskopshagen 1:16 samt de efter verkets tidigare förvärv i enskild ägo kvarvarande delarna av fastigheterna Biskopshagen 1:5, 1:7 och 1:20. Områdets omfattning framgår av bifogade karta samt ingår helt i den av Eders Kungl. Maj:t den 24 januari 1969 fastställda stadsplanen för delar av Biskopshagen, Skällåkra samt Lingome i Varbergs kommun (förutvarande Värö kommun) i Hallands län.

Ägare till berörda fastigheter är:

Biskopshagen 1:5	John Einar Berntsson Biskopshagen, 430 22 Väröbacka
Biskopshagen 1:7	Karl Oskar Larsson och hans hustru Linnea Biskopshagen, 430 22 Väröbacka
Biskopshagen 1:16	Ingvar Karlsson Biskopshagen, 430 22 Väröbacka
Biskopshagen 1:20	Selma Alfrida Larsson och Nils Leonard Larsson Biskopshagen, 430 22 Väröbacka

Äganderättsbevis och jordregisterutdrag bifogas.

I den av Västerbygdens vattendomstol meddelade deldomen den 19 februari 1969 innefattades även verkställighetstillstånd. Denna deldom liksom Vattenöverdomstolens dom den 16 december 1969 har överklagats av enskilda sakägare och ärendet behandlas för närvarande av Högsta domstolen.

Västerbygdens vattendomstol uppsköt i sin deldom prövningen av skumbalkens slutliga utformning samt frågan om tillstånd till utsläpp av avloppsvatten från aggregat tre och fyra. En redovisning i detta hänseende är under utarbetande och kommer att inges till vattendomstolen under november månad 1971.

I handläggningen av detta ärende har deltagit, förutom undertecknad generaldirektör, tekniske direktör Wivstad och byråchef André, den sistnämnde föredragande.

Underdånigst

Jonas V Norrby

Jonas V Norrby

Bertil André　　　　Ingvar Wivstad

Bertil André　　　　Ingvar Wivstad

Stockholm den 9.11.1971

Vidimeras: Margareta Rosenqvist

STATENS VATTENFALLSVERK
VATTENFALL

BAND- OCH KAPITELINDELNING

KAPITEL 1

Introduction and summary

TABLE OF CONTENTS

SECTION 1

INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This Preliminary Safety Analysis Report is submitted in support of an application by the Swedish State Power Board for a permit to construct a nuclear power station designated as Ringhals Power Station, unit 3, Sweden.

The station is to include a pressurized water reactor nuclear steam supply system furnished by Westinghouse Electric Corporation and is similar in design concept to several projects recently licensed or currently under review by the Atomic Energy Commission in the United States of America. The nuclear auxiliary systems will be furnished by Westinghouse Electric Corporation and Monitor. The balance of the station will be designed and constructed by the Applicant, Swedish State Power Board.

The reactor is designed for a warranted power output of 2783 MWt, which is the application rating, with an equivalent station net electrical output of 920 MWe. All safety systems, including containment and engineered safety features, are designed and evaluated for operation at the rated power level.

The remainder of Section 1 of this report summarizes the principal design features and safety criteria of the nuclear unit.

The reactor coolant system consists of three loops, each loop having components (steam generator, pumps, and piping) similar to those for other Westinghouse Nuclear units.

Section 2 contains a description and evaluation of the Ringhals and environs, and supports the suitability of the site for a reactor of the size and type described. Sections 3 and 4 describe the reactor and the reactor coolant system. Section 5 describes the containment structure and related systems. Sections 6 through 11 describe the other auxiliary systems. Sections 5, 6, 7, 8 and 9 include descriptions of the various systems directly related to safeguards. Section 12 reviews the current organization and technical competence of the Applicant and provides preliminary information relating to station organization and personnel training. Section 13 describes, in preliminary form, the approach to the subject of initial tests and operation by the Applicant. Section 14 relates to safety evaluation, and summarizes typical analyses which demonstrate the adequacy of the reactor protection system, the containment and the engineered safety features.

This report presents descriptive material and analyses of a preliminary design. As the station design progresses to a final detailed design, the station description, preliminary design numbers and the related analyses will be subject to change and refinement.

1.1.1 Fuel Assembly Design

The fuel assembly will incorporate the rod cluster control concept in a canless 15 x 15 fuel and control rod array, using Inconel spring clip grids to provide support for the fuel rods. Extensive out-of-pile tests have been performed on this concept; successful in-pile tests have been performed in the Saxton reactor; and operating experience has been obtained in the San Onofre and Connecticut Yankee plants.

The fuel rod design will use Zircaloy-4 as a clad material, which has proved successful in the Carolinas Virginia Test Reactor and Saxton reactors and Yankee test assemblies, and is scheduled for eventual use in all Westinghouse reactors under construction or being currently proposed for construction.

Operation at 2783 MWt will yield a maximum steady state thermal power of 18.8 kw/ft. These values are justified by the results of in-core experiments by Westinghouse and others at these and higher specific power ratings.

1.1.2 Containment

The reactor containment concept is based upon the use of a prestressed concrete containment structure. Following the postulated loss-of-coolant accident, described in Section 14, the containment is depressurized by the use of redundant spray cooling systems, thereby terminating outleakage to the environment within 48 hours after initiation of the accident.

1.1.3 Conclusions

The pressurized water reactor unit of the type to be used at Ringhals possesses inherent stability characteristics and safety features. These are typified by the Doppler coefficient effect in the fuel and the barriers against fission product release. Engineered safety features and safeguard systems, such as the safety injection system and the containment spray system complement these characteristics.

The Applicant submits that Ringhals can be constructed and operated without hazard to the general public, because of the engineered safeguards described herein and the analyses that ensure safety to the public and operating personnel.

1.2 SUMMARY - STATION DESCRIPTION

1.2.1 General

Ringhals No. 3 incorporates a closed-cycle pressurized water nuclear steam supply system, two turbine generators, and their necessary auxiliaries. Radioactive waste disposal systems, a fuel handling system, and all auxiliaries, structures, and other on-site facilities required for a complete and operable nuclear power station are also provided. The general arrangement of the units is shown in the site plan, figure 2:1 and the plot plan, figure 2:2.

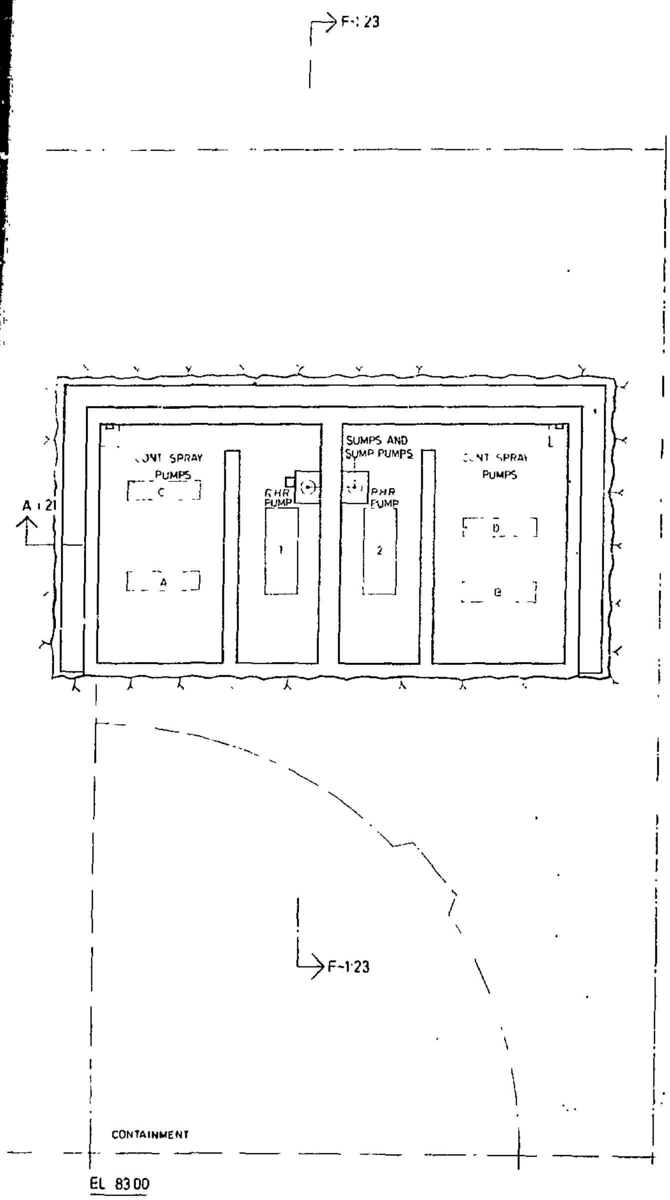
1.2.2 Site

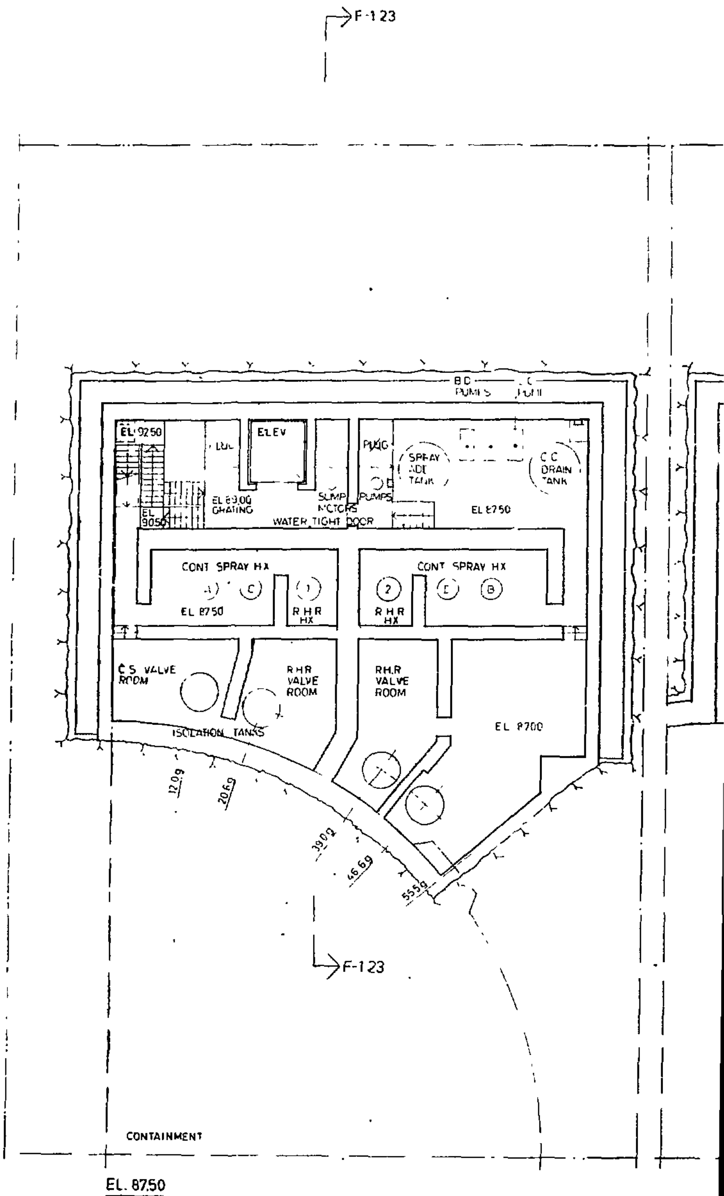
Data rörande förläggningsplatsen och dess omgivningar har redovisats i koncessionsansökan av den 10 juni 1970. Föredragningar har skett inför reaktorförläggningskommittén. Kommittén kommer fortsättningsvis att delges resultat från pågående undersökningar på platsen. I avsnitt 2 redovisas de tillkommande förhållanden som bedömes vara av intresse.

1.2.3 Structures

The major structures are the reactor containment, auxiliary building, fuel building, turbine building, and service building which includes the main control area. The general arrangement of the reactor containment, auxiliary building, and fuel building, showing typical layouts, are given on Figs. 1.11-1.16, 1.21-1.24 and 1.31-1.32.

The reactor containment is a steel-lined, prestressed concrete cylinder with a hemispherical dome and a flat reinforced concrete foundation mat. The containment is designed to withstand the internal pressure accompanying the design basis accident (DBA), is virtually leaktight, and provides adequate radiation shielding for both normal operation and conditions. There is no outleakage of activity from the containment structure when it is at subatmospheric pressure.


F-1 23
A 1 21
CONT SPRAY PUMPS
C
A
RHR PUMP
1
SUMPS AND SUMP PUMPS
2
CONT SPRAY PUMPS
D
B
F-1 23
CONTAINMENT
EL 83.00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ELEV
SPRAY ADD TANK
CC DRAIN TANK
EL 89.00 GRATING
SUMP MOTORS
PUMPS
EL 90.50
WATER TIGHT DOOR
EL 87.50
CONT SPRAY HX
A
C
1
EL 87.50
RHR HX
CONT SPRAY HX
2
D
B
RHR HX
CS VALVE ROOM
RHR VALVE ROOM
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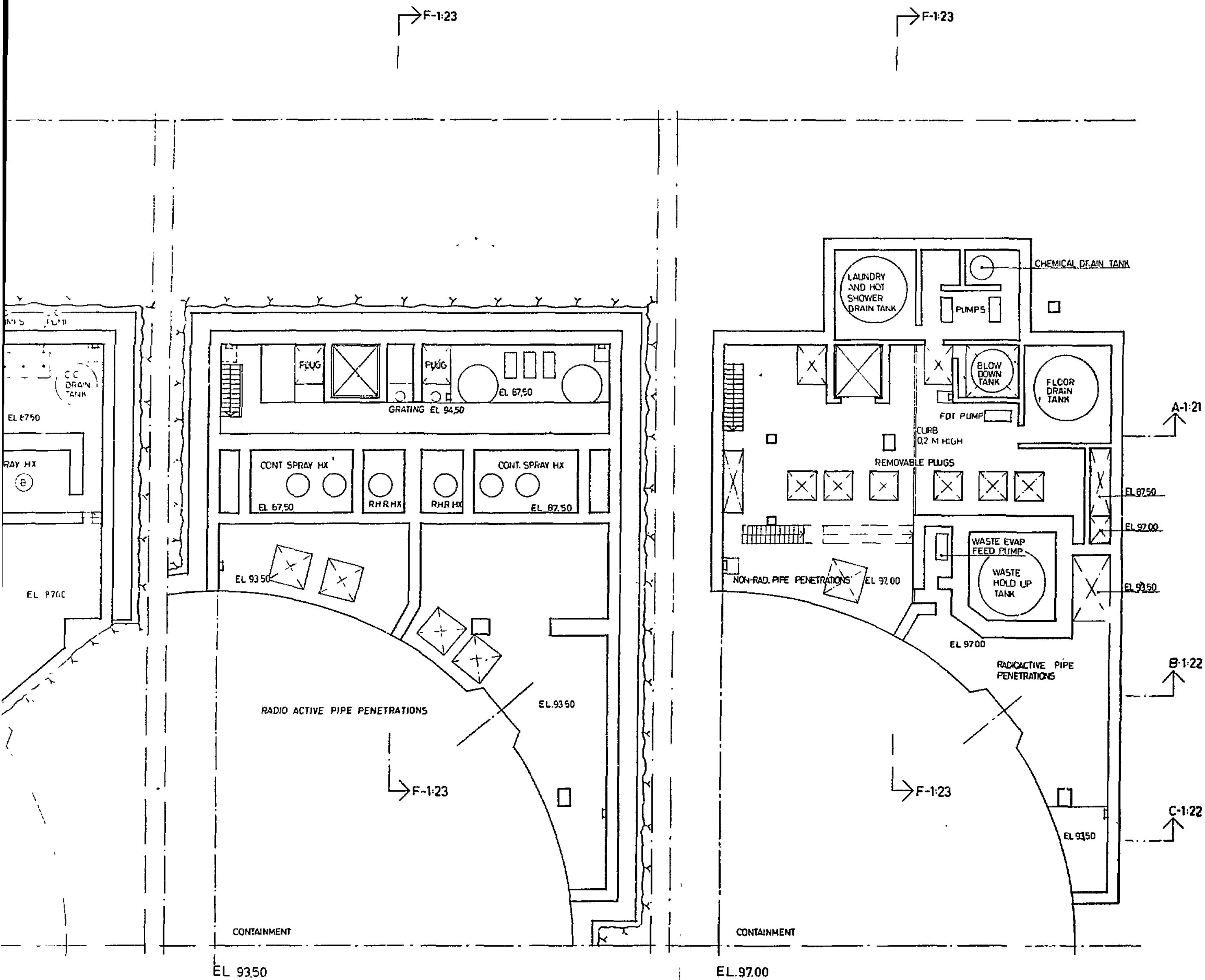
EL 87.00
ISOLATION TANKS
F-1 23
CONTAINMENT
EL. 87.50



WESTINGHOUSE PROPRIETARY DATA

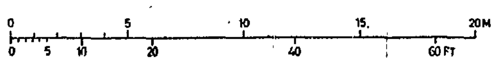
WESTINGHOUSE
PROPOSAL
ALTERNATIVE No 2

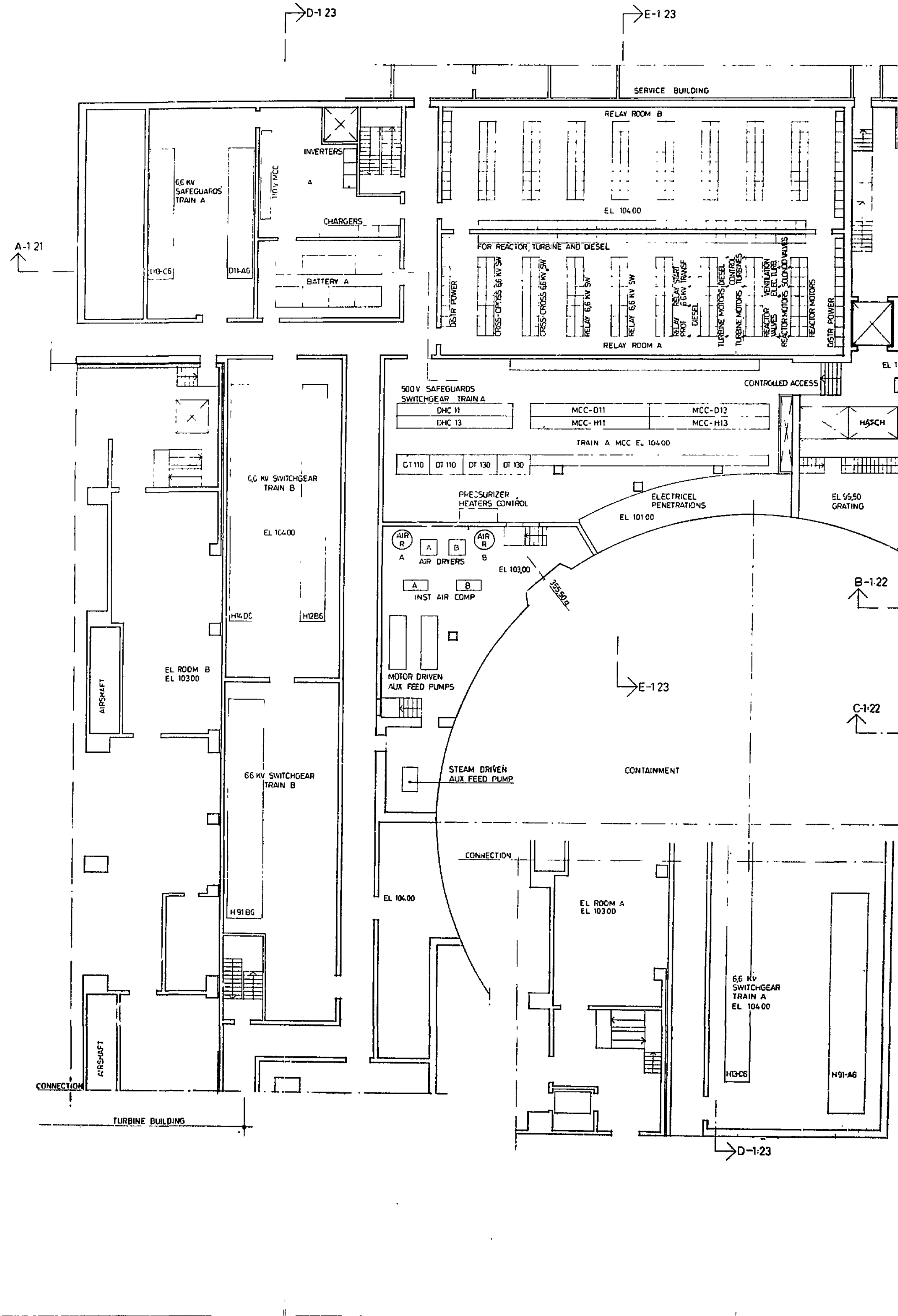
NUCLEAR POWER STATION
AUXILIARY BUILDING
PLAN AT EL 83.00, 87.50, 91.00, 97.00

			14290
1:100		1:11	

MARCH 15, 1971

VBB





D-1 23
E-1 23
SERVICE BUILDING
RELAY ROOM B
EL 104.00
FOR REACTOR, TURBINE AND DIESEL
RELAY ROOM A
INVERTERS
CHARGERS
BATTERY A
6,6 KV SAFEGUARDS TRAIN A
A-1 21
500 V SAFEGUARDS SWITCHGEAR TRAIN A
DHC 11
DHC 13
MCC-D11
MCC-H11
MCC-D13
MCC-H13
TRAIN A MCC EL 104.00
CONTROLLED ACCESS
HATCH
6,6 KV SWITCHGEAR TRAIN B
EL 104.00
PRESSURIZER HEATERS CONTROL
ELECTRICEL PENETRATIONS
EL 101.00
EL 96.50 GRATING
AIR DRYERS
INST AIR COMP
EL 103.00
B-1:22
EL ROOM B EL 103.00
AIRSHAFT
MOTOR DRIVEN AUX FEED PUMPS
E-1 23
C-1:22
66 KV SWITCHGEAR TRAIN B
STEAM DRIVEN AUX FEED PUMP
CONTAINMENT
CONNECTION
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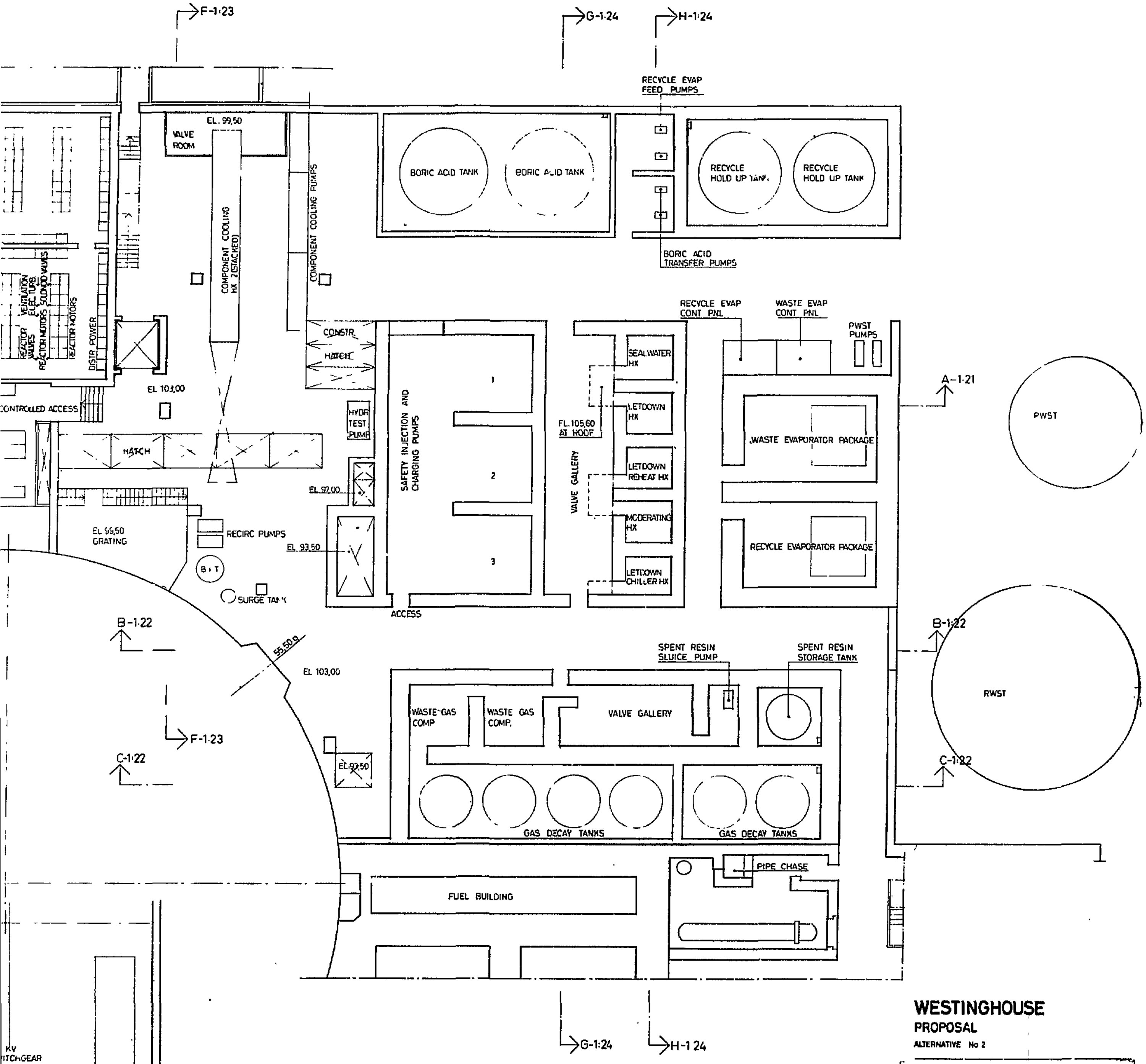
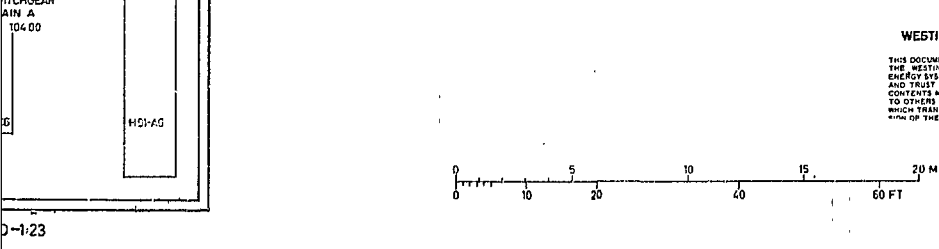
EL ROOM A EL 103.00
6,6 KV SWITCHGEAR TRAIN A EL 104.00
AIRSHAFT
CONNECTION
TURBINE BUILDING
D-1:23





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WESTINGHOUSE
PROPOSAL
ALTERNATIVE No 2

NUCLEAR POWER STATION
AUXILIARY BUILDING
PLAN AT EL 103,00 AND 104,00

14290
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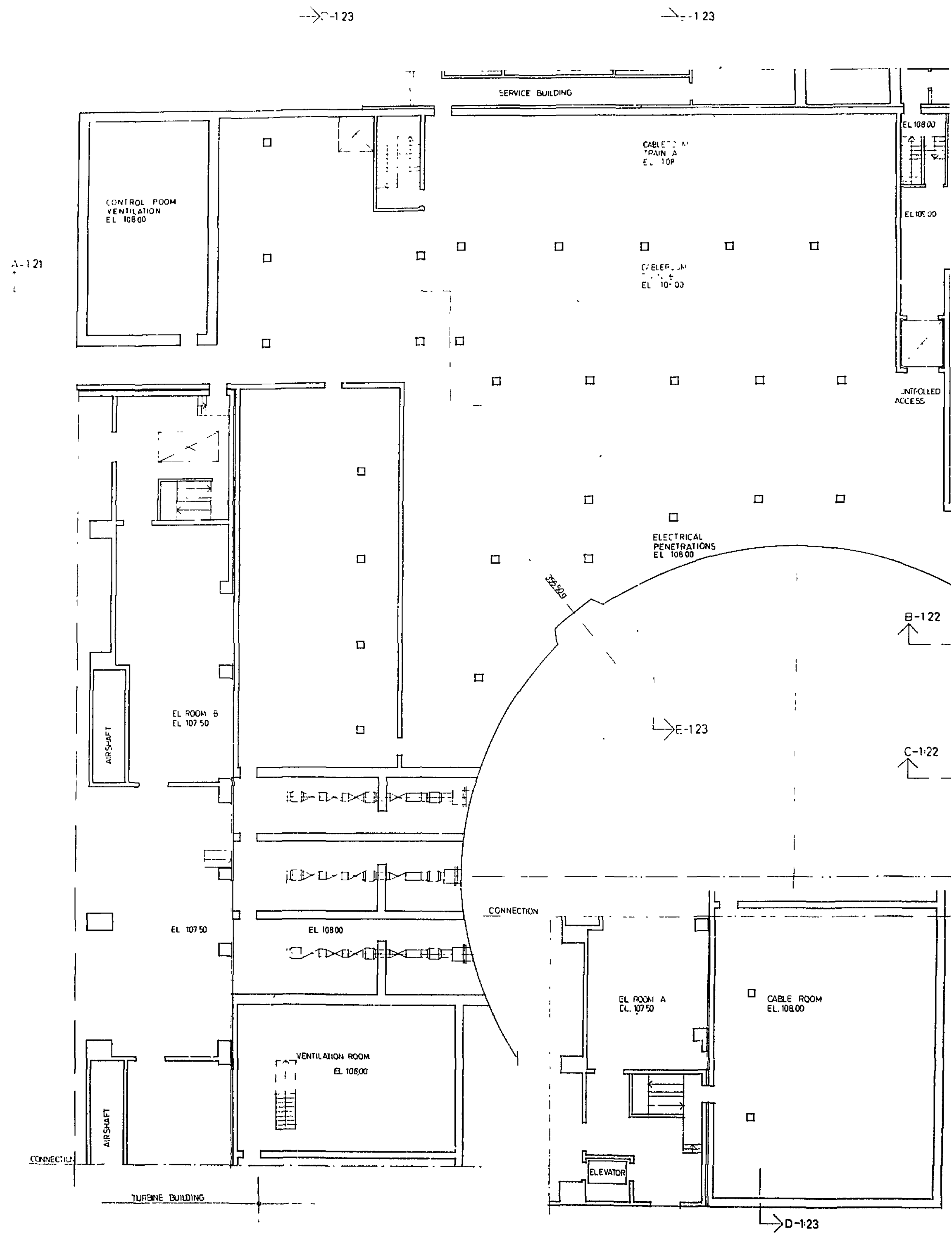
1:12
MARCH 15, 1971
VBB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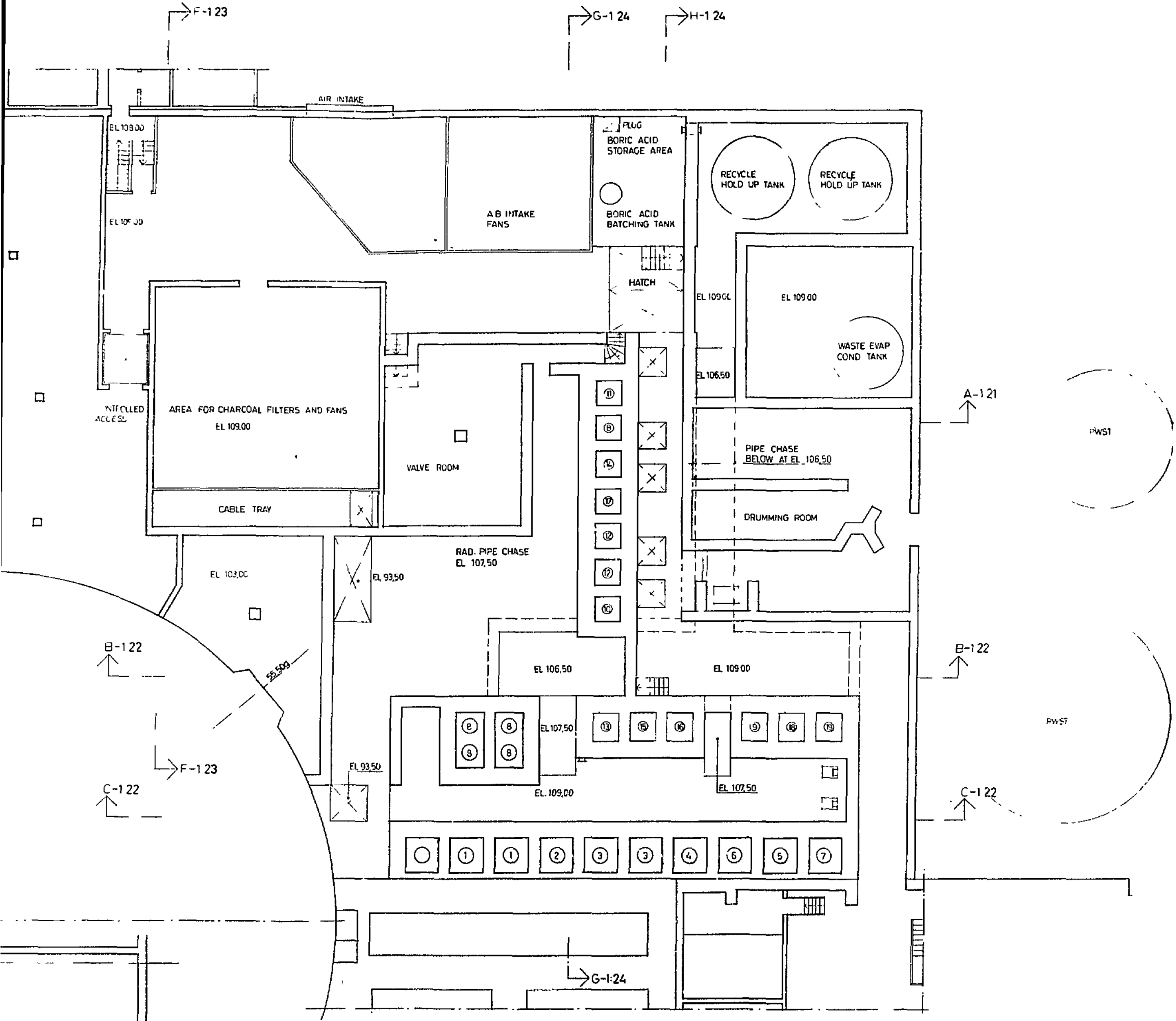
CONTROL ROOM VENTILATION EL 108.00
A-1 21
EL 108.00
EL 105.00
ELECTRICAL PENETRATIONS EL 108.00
B-1 22
E-1 23
C-1 22
EL ROOM B EL 107.50
AIRSHAFT
EL 107.50
EL 108.00
CONNECTION
EL ROOM A EL. 107.50
CABLE ROOM EL. 108.00
VENTILATION ROOM EL 108.00
ELEVATOR
AIRSHAFT
TURBINE BUILDING
D-1 23



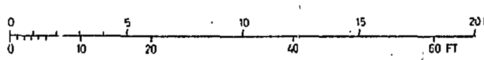
DEMINERALIZERS

1. MIXED BED
2. CATION BED
3. RECYCLE EVAP. FEED
4. RECYCLE EVAP CONDENSATE
5. WASTE EVAP. CONDENSATE
6. SPENT FUEL PIT
7. WASTE MONITOR
8. BORON THERMAL REGENERATION ION HX

FILTERS

8. BORIC ACID
9. SPENT RESIN SLUICE
10. R.C.
11. SEAL WATER
12. SEAL WATER INJ
13. CONC RECYCLE
14. WASTE EVAP
15. RECYCLE EVAP
16. RECYCLE CONDENSATE
17. WASTE CONDENSATE
18. FLOOR DRAIN TANK
19. LAUNDRY AND HOT SHOWER TANK

H-1:24



WESTINGHOUSE PROPRIETARY DATA

WESTINGHOUSE
PROPOSAL
ALTERNATIVE No 2

NUCLEAR POWER STATION
AUXILIARY BUILDING
PLAN AT EL. 107,50

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	ME		14290

SKALA		REV.
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MARCH 15, 1971

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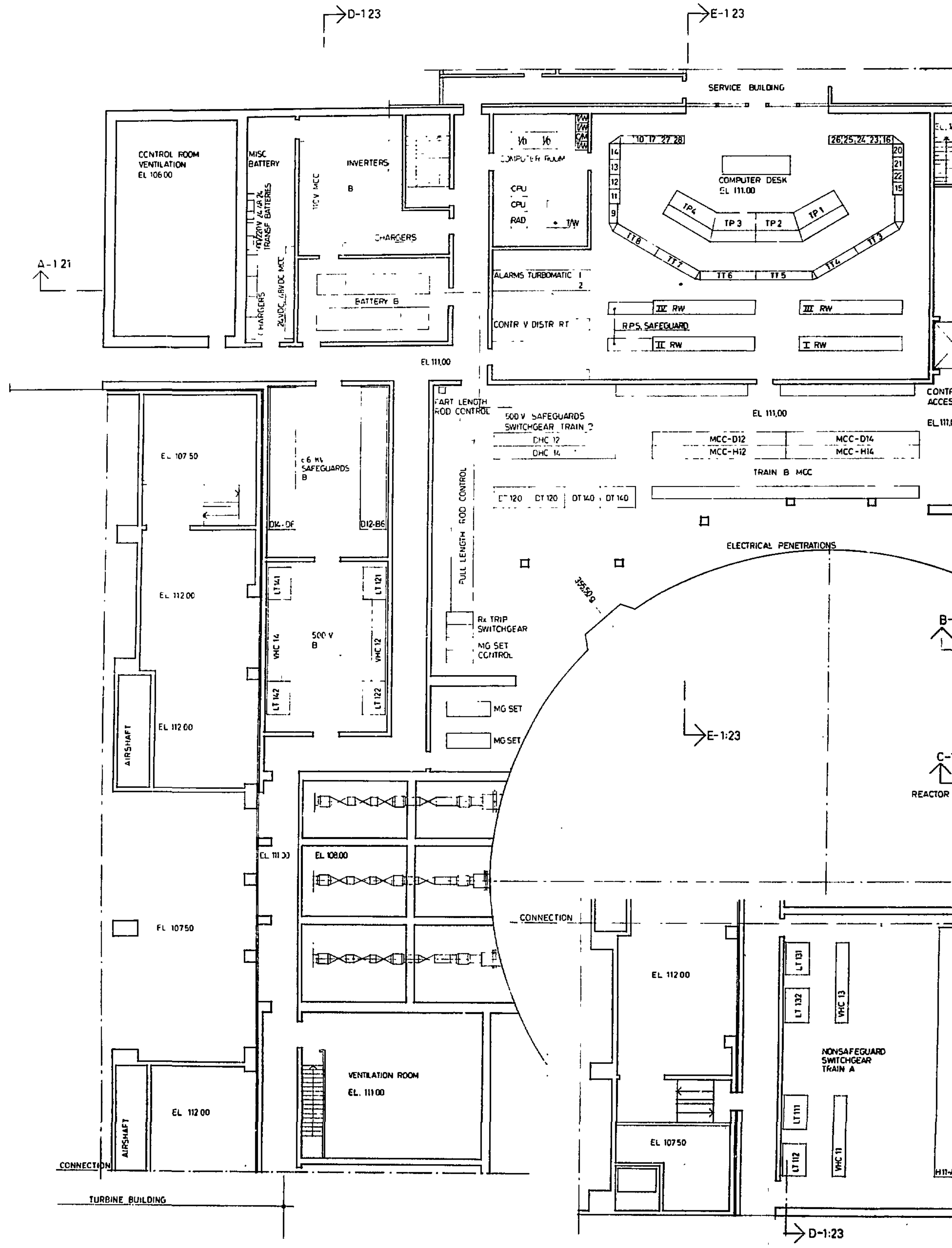
VATTENBYGGNADSBYRÅN

ROOM

23



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E-1 23
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INVERTERS
B
CHARGERS
BATTERY B
COMPUTER ROOM
CPU
CPU
RAD
COMPUTER DESK EL 111.00
TP 1
TP 2
TP 3
TP4
ALARMS TURBOMATIC
CONTR V DISTR RT
RPS SAFEGUARD
IV RW
III RW
II RW
I RW
A-1 21
EL 111,00
PART LENGTH ROD CONTROL
500 V SAFEGUARDS SWITCHGEAR TRAIN 2
DHC 12
DHC 14
MCC-D12
MCC-H12
MCC-D14
MCC-H14
TRAIN B MCC
ELECTRICAL PENETRATIONS
FULL LENGTH ROD CONTROL
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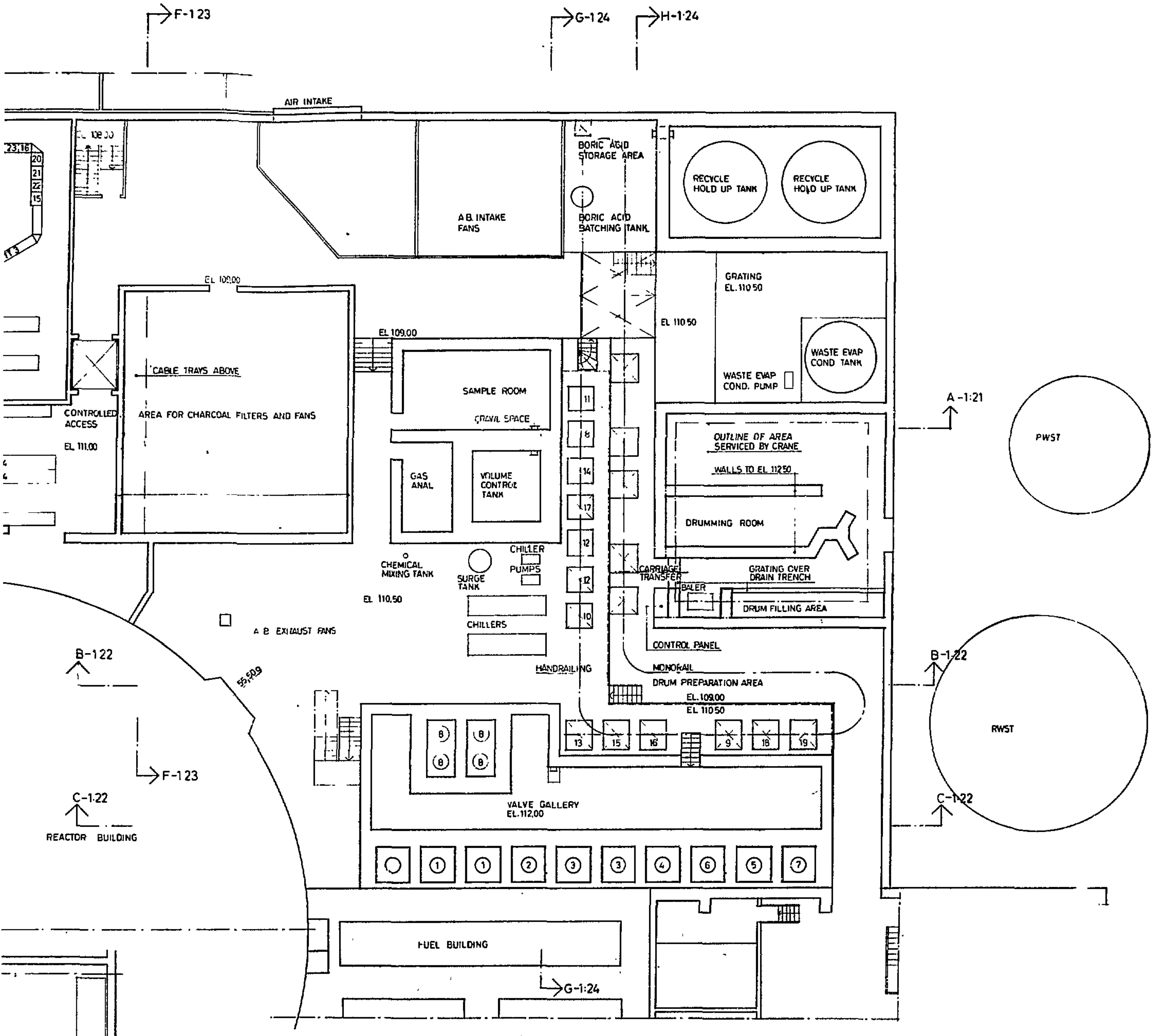
6 kV SAFEGUARDS B
D14-DF
D12-B6
EL 112 00
LT 141
LT 121
VHC 14
500 V B
VHC 12
LT 142
LT 122
AIRSHAFT
Rx TRIP SWITCHGEAR
MG SET CONTROL
MG SET
MG SET
E-1:23
REACTOR
EL 108.00
CONNECTION
EL 10750
EL 112 00
LT 131
LT 132
VHC 13
NONSAFEGUARD SWITCHGEAR TRAIN A
LT 111
LT 112
VHC 11
VENTILATION ROOM EL. 111 00
EL 10750
CONNECTION
TURBINE BUILDING
D-1:23



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2 CATION BED.
3 RECYCLE EVAP. FEED
4 RECYCLE EVAP. CONDENSATE
5 WASTE EVAP. CONDENSATE
6. SPENT FUEL PIT
7. WASTE MONITOR
8. BORON THERMAL REGENERATION ION HX

FILTERS
8. BORIC ACID
9. SPENT RESIN SLUICE
10 R C.
11. SEAL WATER
12. SEAL WATER INJ
13 CONC. RECYCLE
14. WASTE EVAP
15. RECYCLE EVAP. FEED
16. RECYCLE EVAP. CONDENSATE
17. WASTE EVAP CONDENSATE
18. FLOOR DRAIN TANK
19. LAUNDRY AND HOT SHOWER TANK

H-1:24

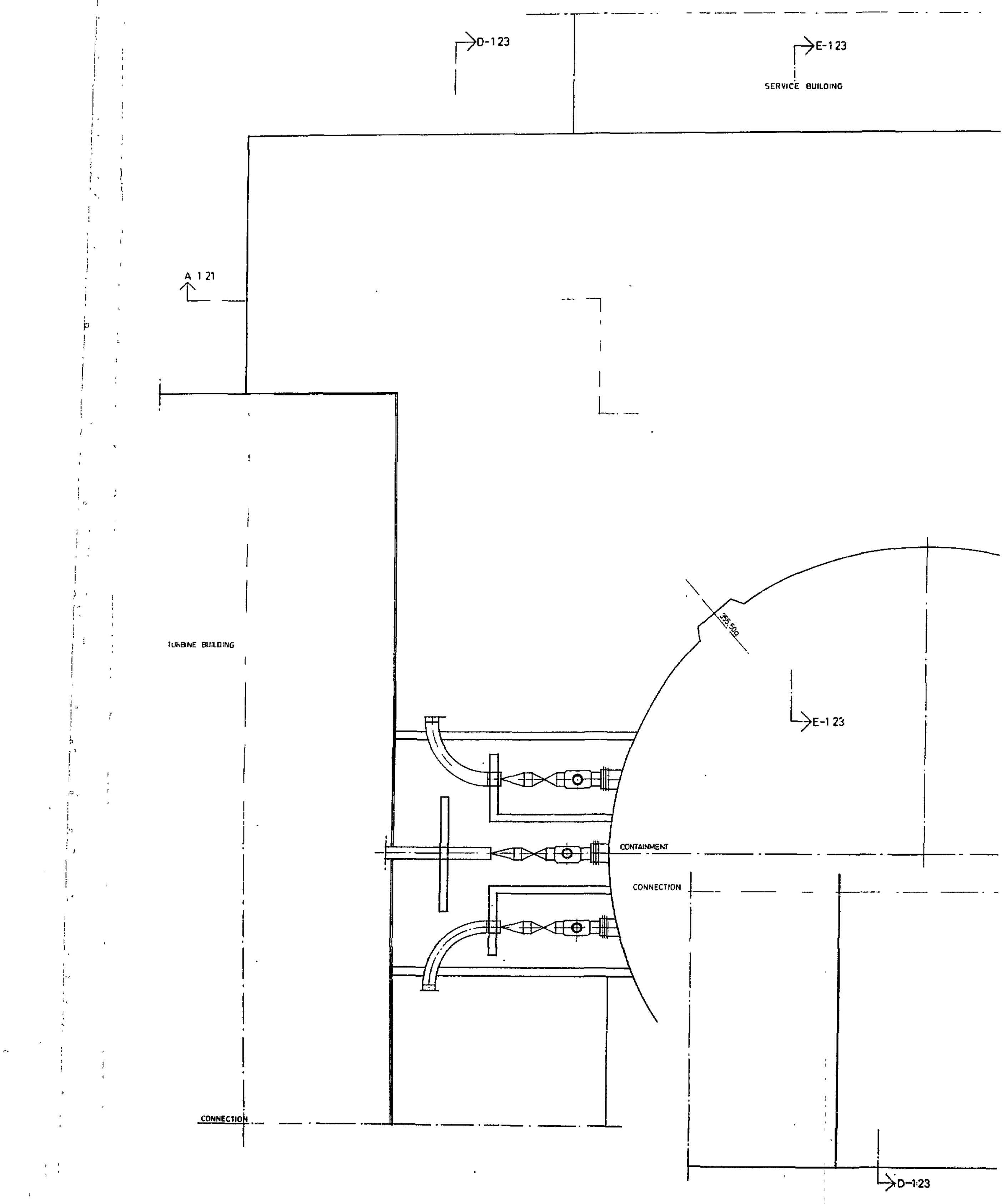
0 5 10 15 20 M
0 10 20 40 60 FT

WESTINGHOUSE PROPRIETARY DATA

WESTINGHOUSE
PROPOSAL
ALTERNATIVE No 2

NUCLEAR POWER STATION
AUXILIARY BUILDING
PLAN AT EL. 109,00 AND 111,00

HSE
14290
1:100
1:14
MARCH 15, 1971
VBB



D-123
E-123
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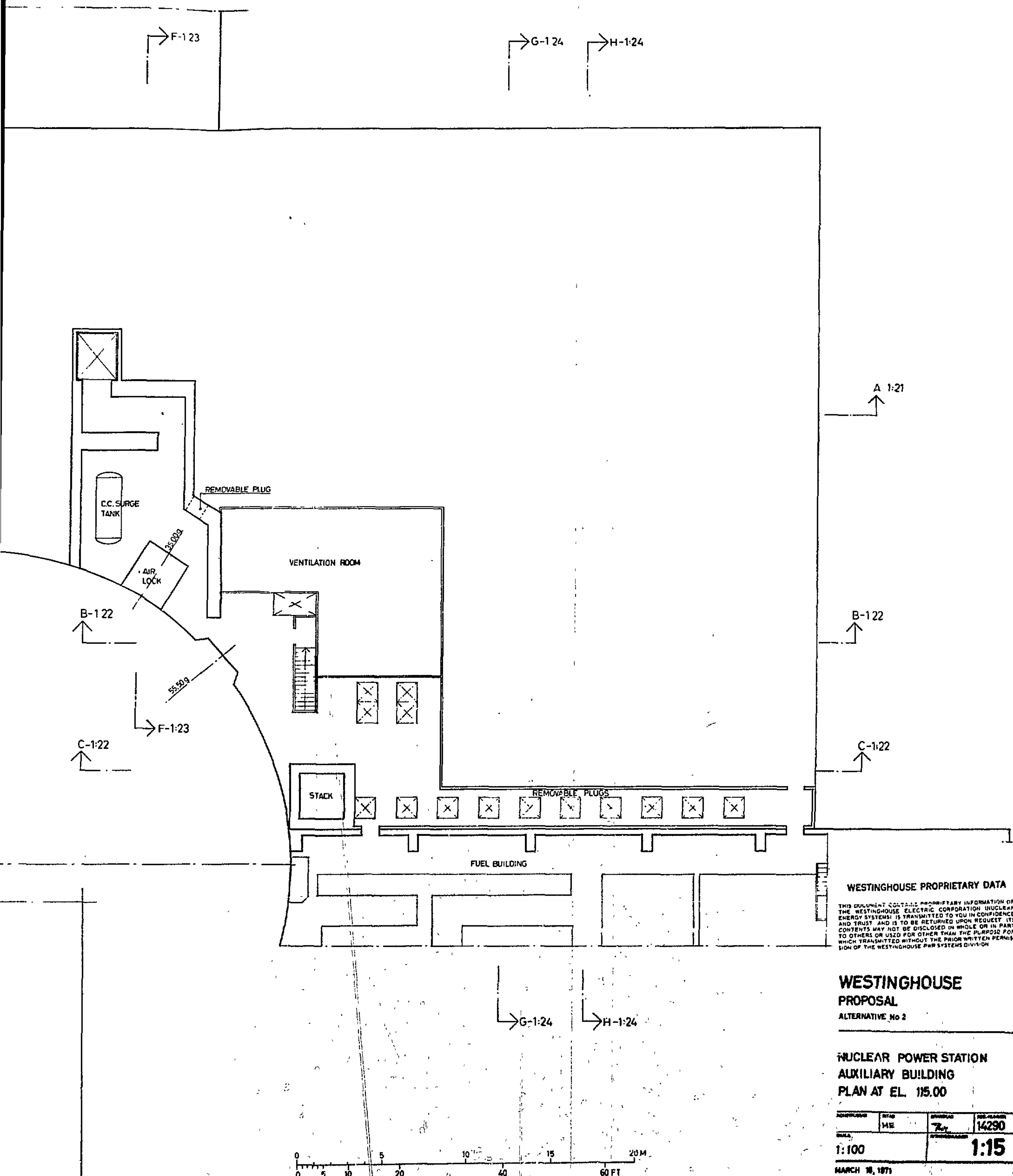
A 121
TURBINE BUILDING
E-1 23
CONTAINMENT
CONNECTION
CONNECTION
D-1·23



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WESTINGHOUSE
PROPOSAL
ALTERNATIVE No 2

NUCLEAR POWER STATION
AUXILIARY BUILDING
PLAN AT EL. 115.00

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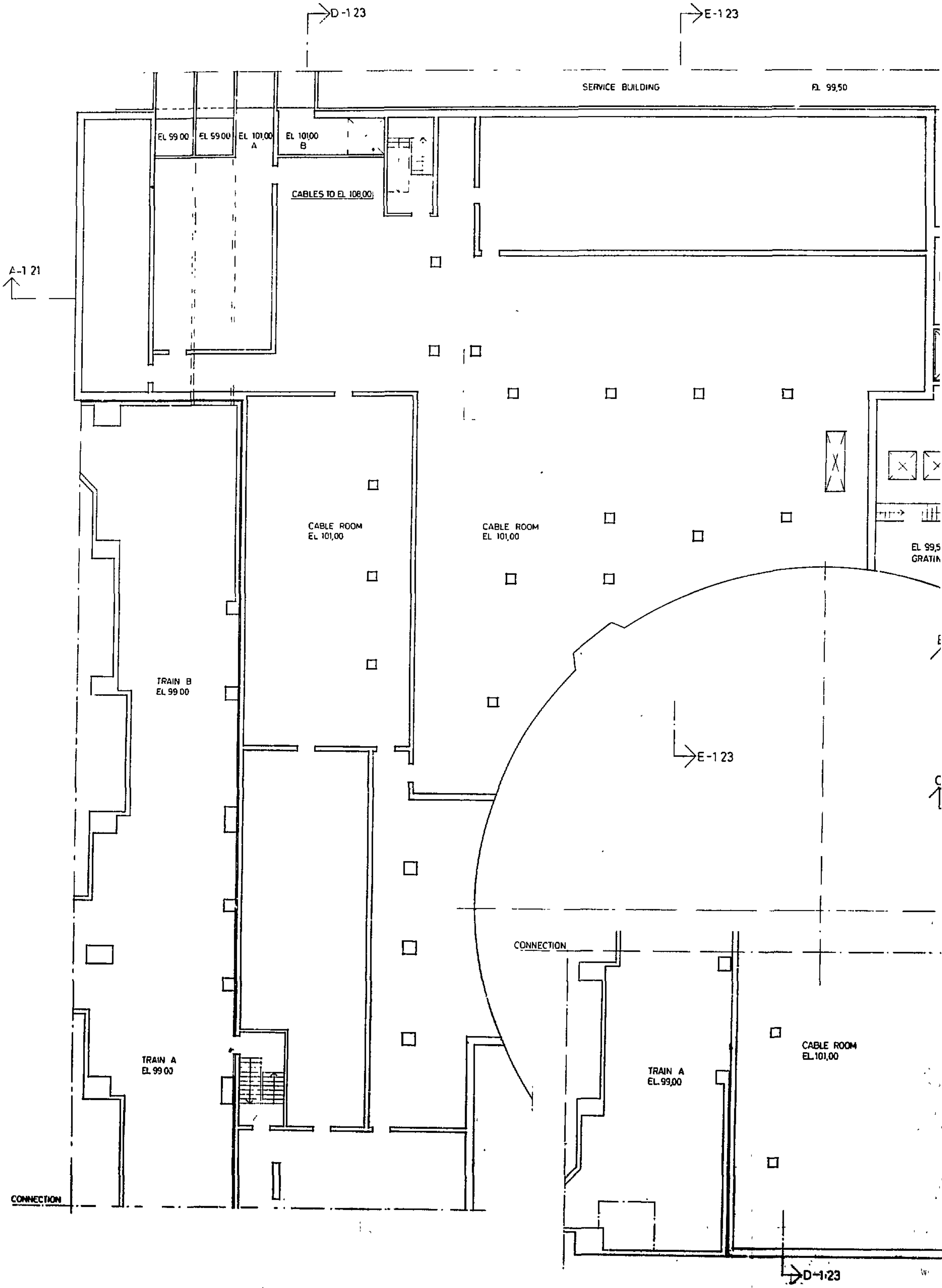
1:100		1:15	

MARCH 18, 1971

VBB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E-1 23
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EL 99.00
EL 101,00 A
EL 101,00 B
CABLES TO EL 108,00
A-1 21
CABLE ROOM EL 101,00
CABLE ROOM EL 101,00
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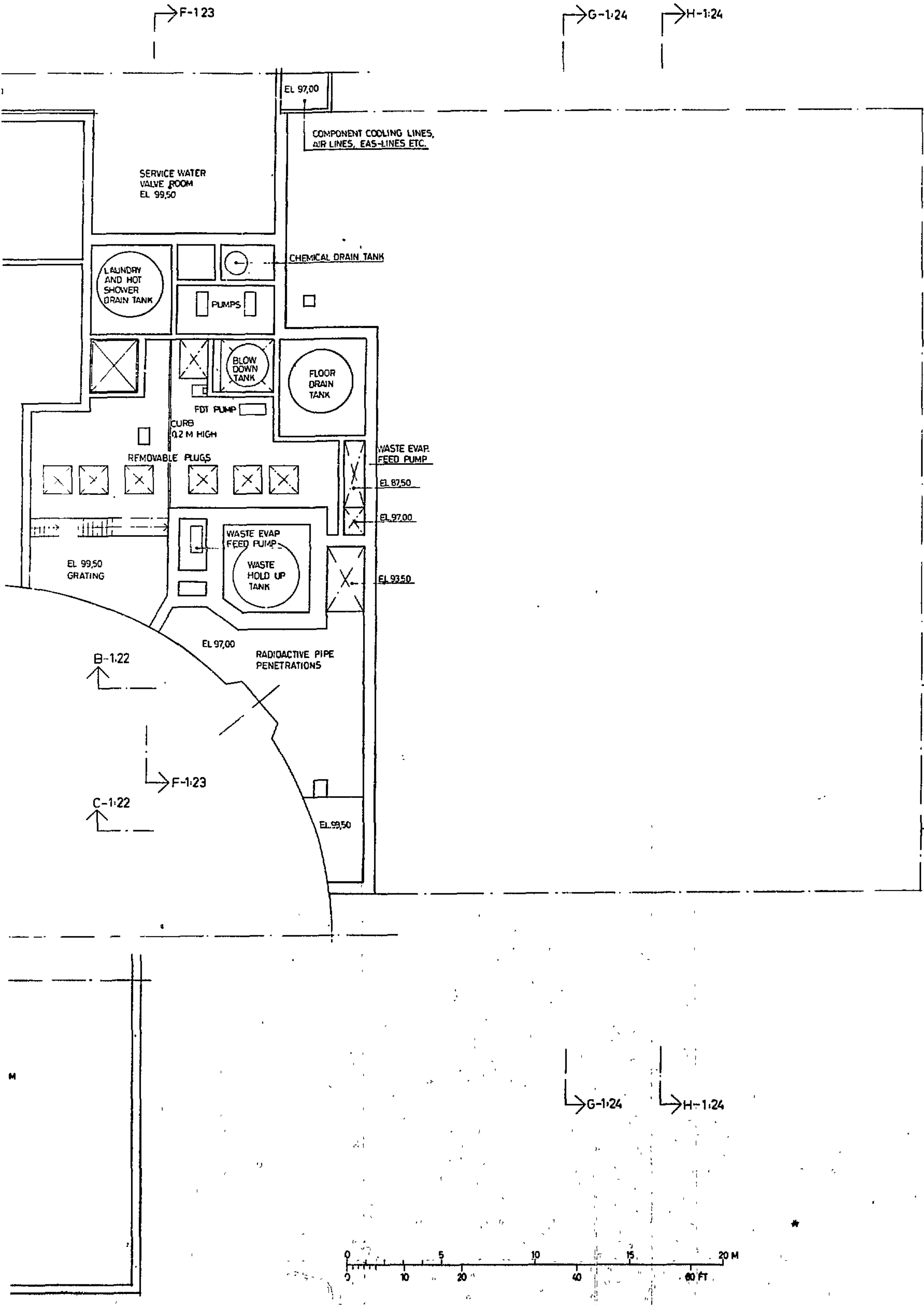
CONNECTION
D-1 23



WESTINGHOUSE
PROPOSAL
ALTERNATIVE No 2

NUCLEAR POWER STATION
AUXILIARY BUILDING
PLAN AT EL.101,00

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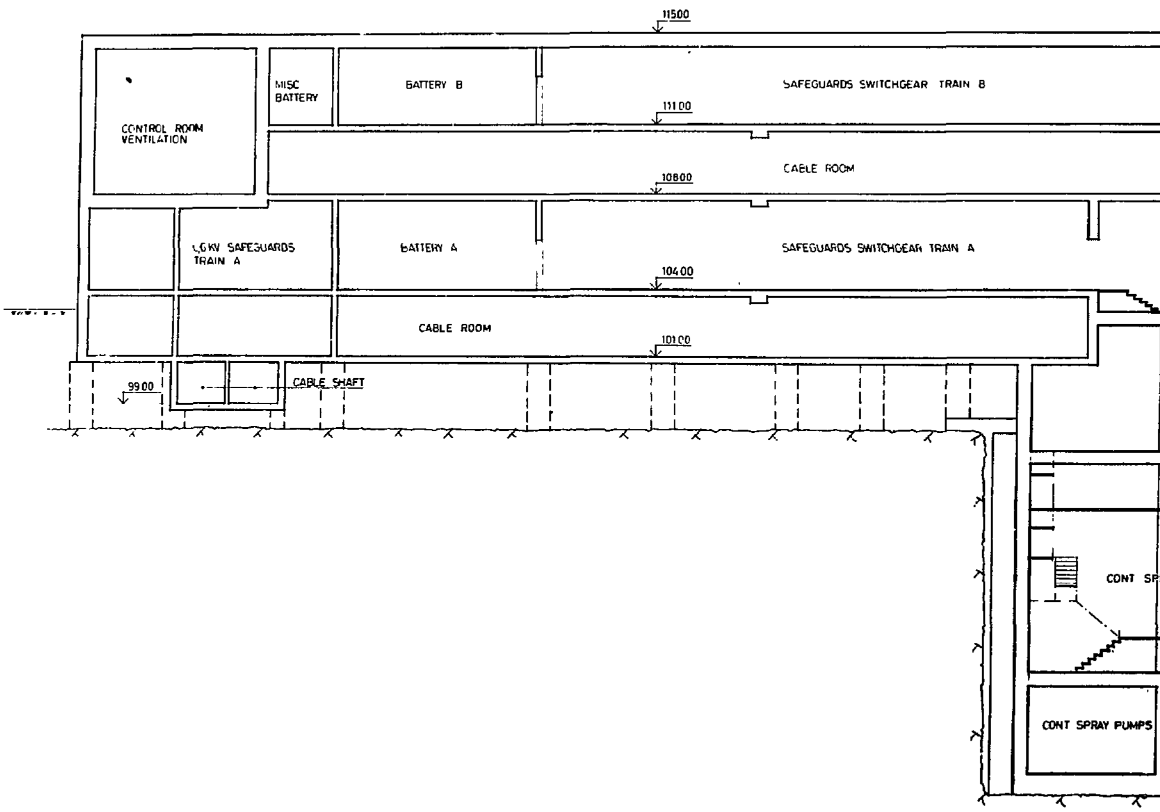
MARCH 15, 1971

VBB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BATTERY B
SAFEGUARDS SWITCHGEAR TRAIN B
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CABLE ROOM
10800
6,6KV SAFEGUARDS TRAIN A
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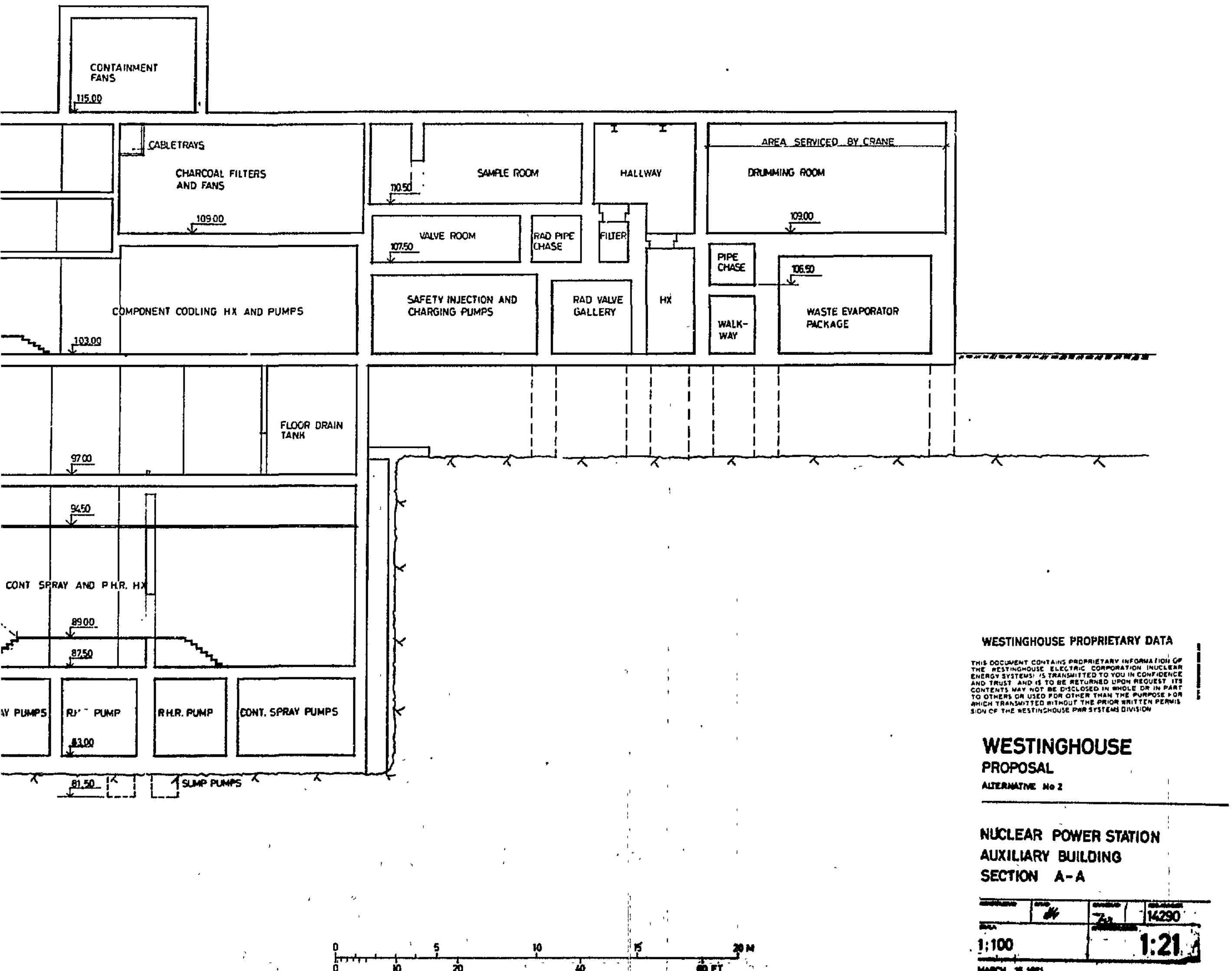
CABLE ROOM
10100
9900
CABLE SHAFT
CONT SP
CONT SPRAY PUMPS



CONTAINMENT FANS
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CHARCOAL FILTERS AND FANS
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110.50
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HALLWAY
AREA SERVICED BY CRANE
DRUMMING ROOM
109.00
VALVE ROOM
107.50
RAD PIPE CHASE
FILTER
PIPE CHASE
106.50
COMPONENT COOLING HX AND PUMPS
103.00
SAFETY INJECTION AND CHARGING PUMPS
RAD VALVE GALLERY
HX
WALK-WAY
WASTE EVAPORATOR PACKAGE
FLOOR DRAIN TANK
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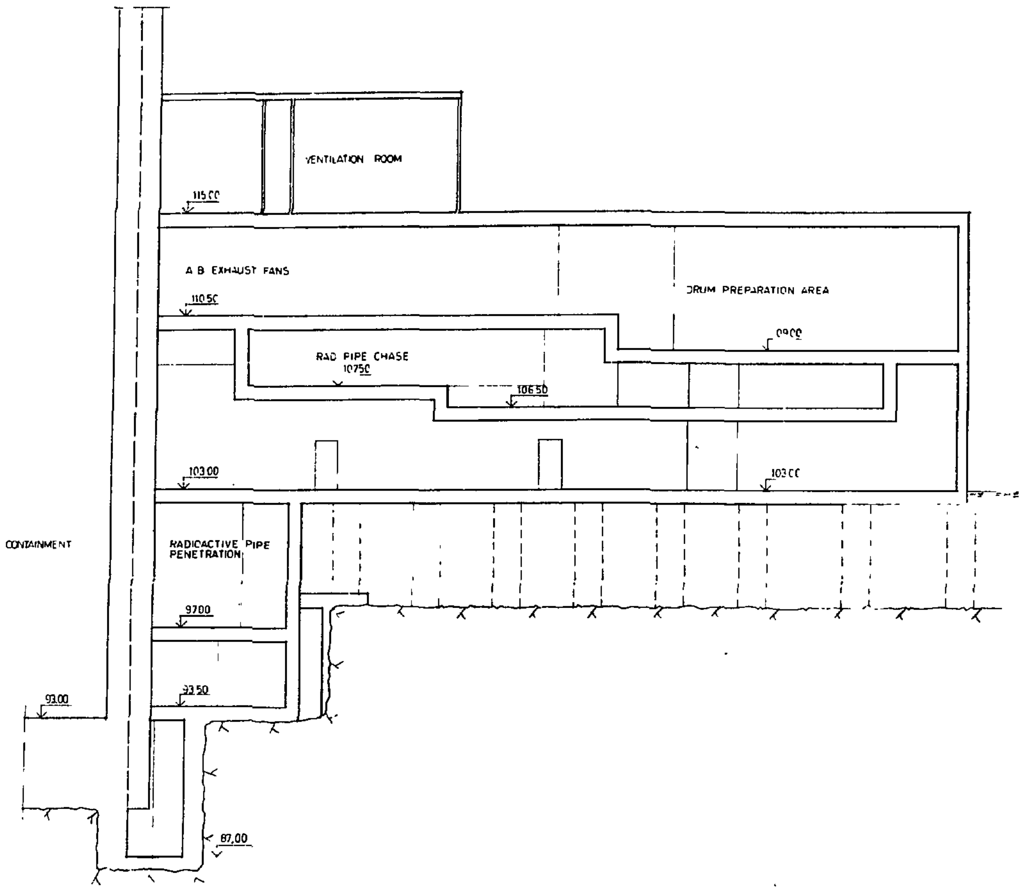
94.50
CONT SPRAY AND P.H.R. HX
89.00
87.50
PUMPS
PUMP
R.H.R. PUMP
CONT. SPRAY PUMPS
83.00
81.50
SUMP PUMPS
0 5 10 15 20 M
0 10 20 40 60 FT
WESTINGHOUSE PROPRIETARY DATA
WESTINGHOUSE
PROPOSAL
ALTERNATIVE No 2
NUCLEAR POWER STATION
AUXILIARY BUILDING
SECTION A-A
14290
1:100
1:21


VENTILATION ROOM
A B EXHAUST FANS
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DRUM PREPARATION AREA
RAD PIPE CHASE
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103.00
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RADIOACTIVE PIPE PENETRATION
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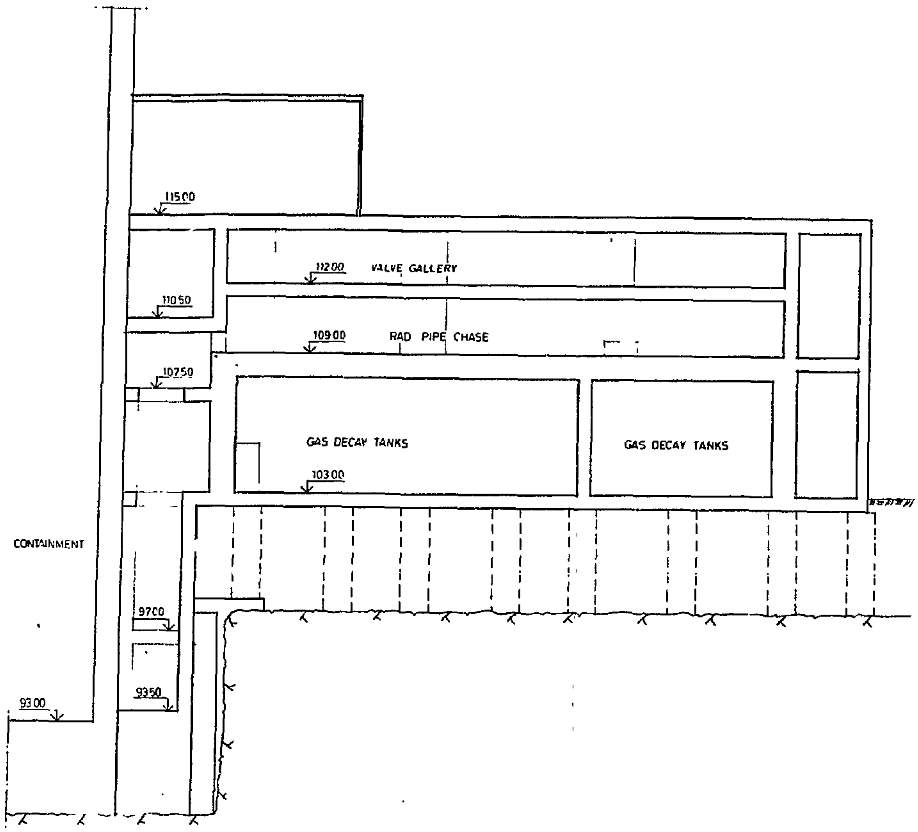
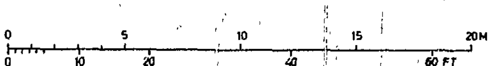
SECTION B-B



SECTION C-C



WESTINGHOUSE PROPRIETARY DATA

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WESTINGHOUSE
PROPOSAL
ALTERNATIVE No 2

NUCLEAR POWER STATION
AUXILIARY BUILDING
SECTION B-B, C-C

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MARCH 16, 1971

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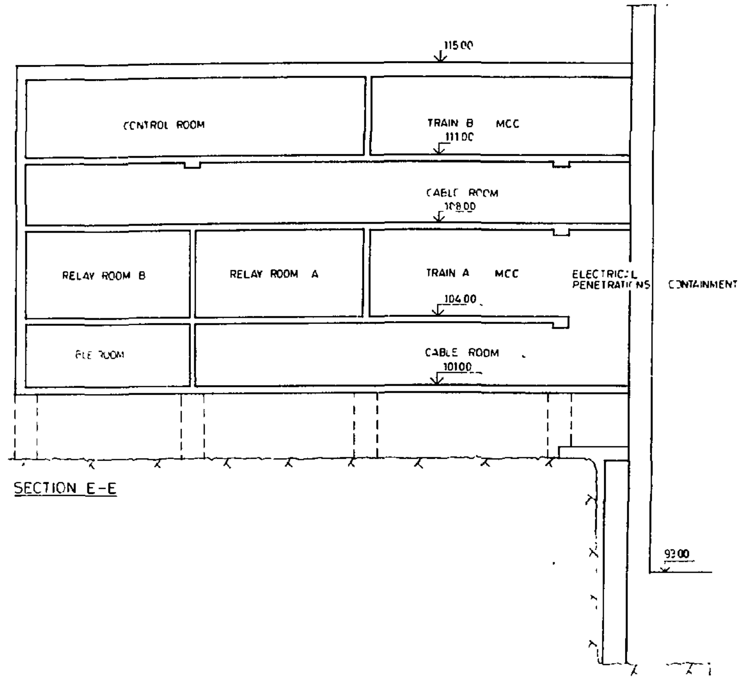

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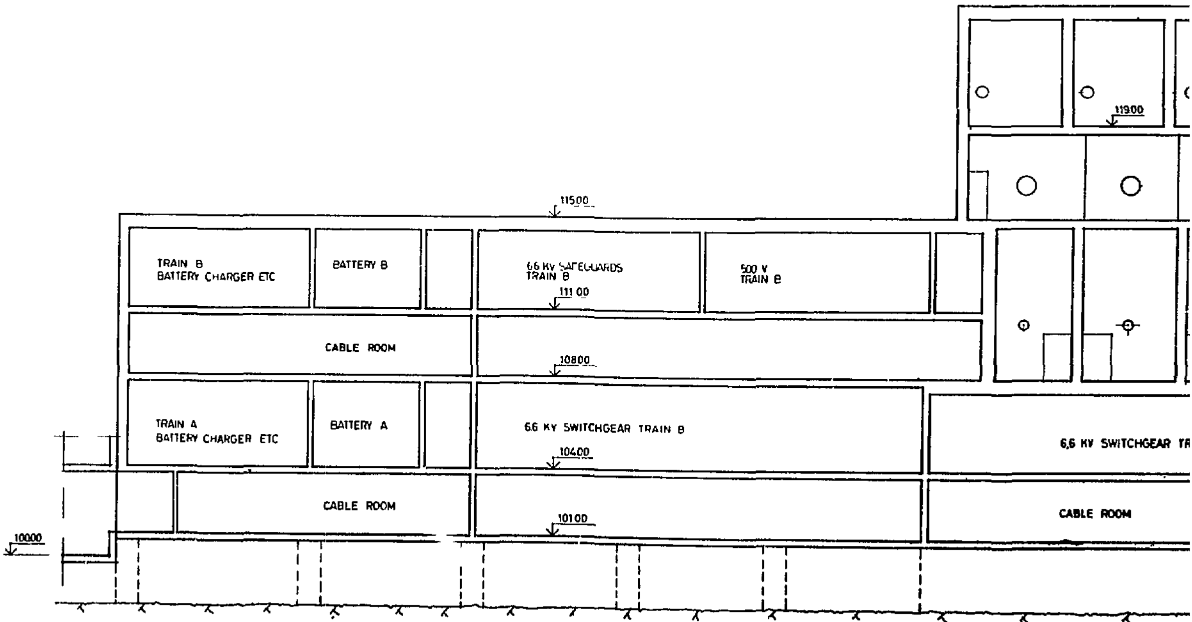

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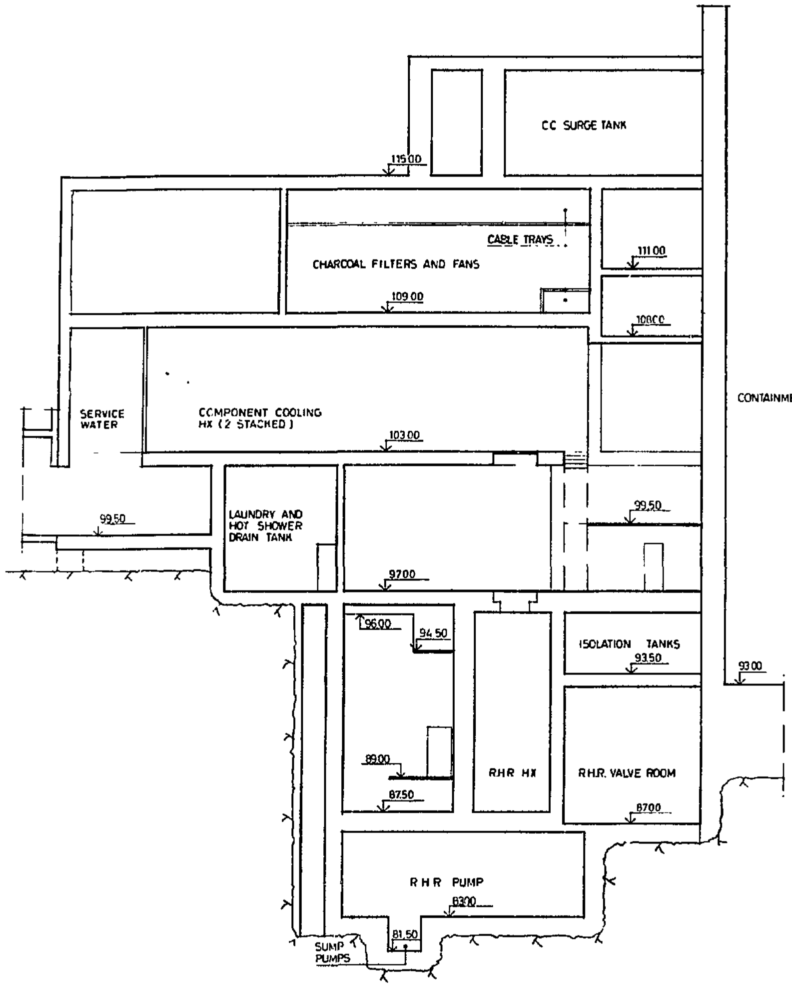

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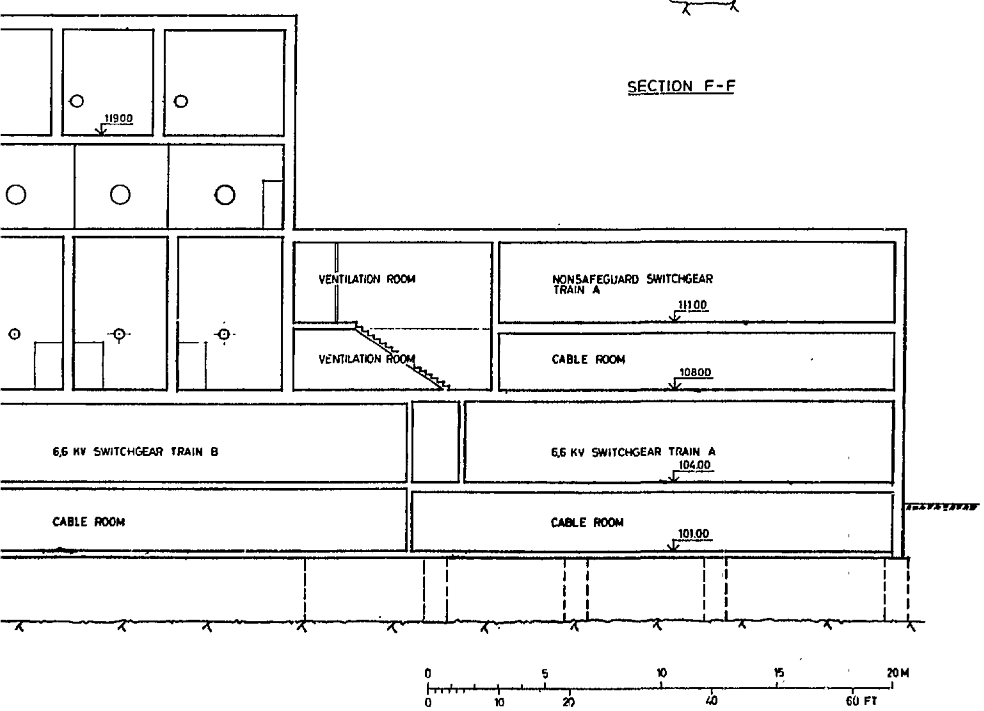

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WESTINGHOUSE
PROPOSAL
ALTERNATIVE No 2

NUCLEAR POWER STATION
AUXILIARY BUILDING
SECTION D-D, E-E, F-F

			14290
1:100		1:23	

MARCH 16, 1971

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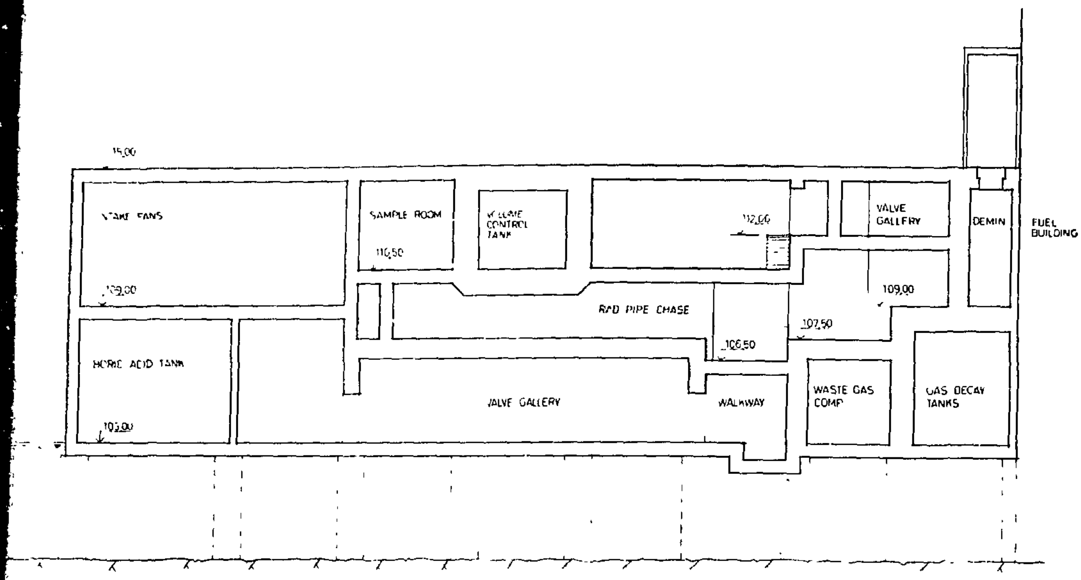

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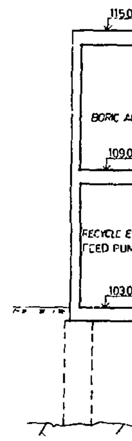

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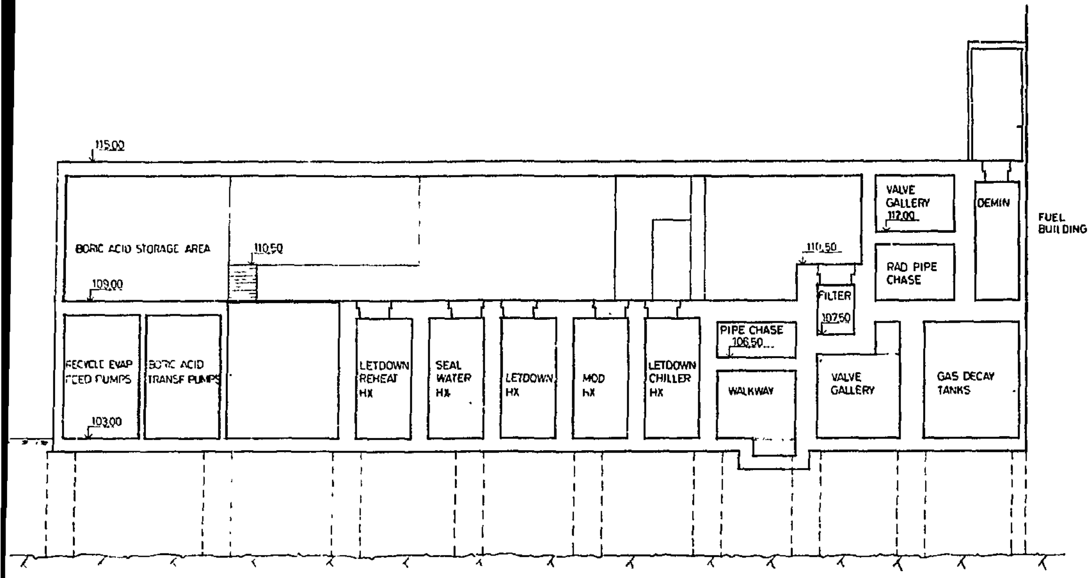

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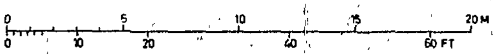

WESTINGHOUSE PROPRIETARY DATA

WESTINGHOUSE
PROPOSAL
ALTERNATIVE No 2

NUCLEAR POWER STATION
AUXILIARY BUILDING
SECTION G-G, H-H

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MARCH 16, 1971

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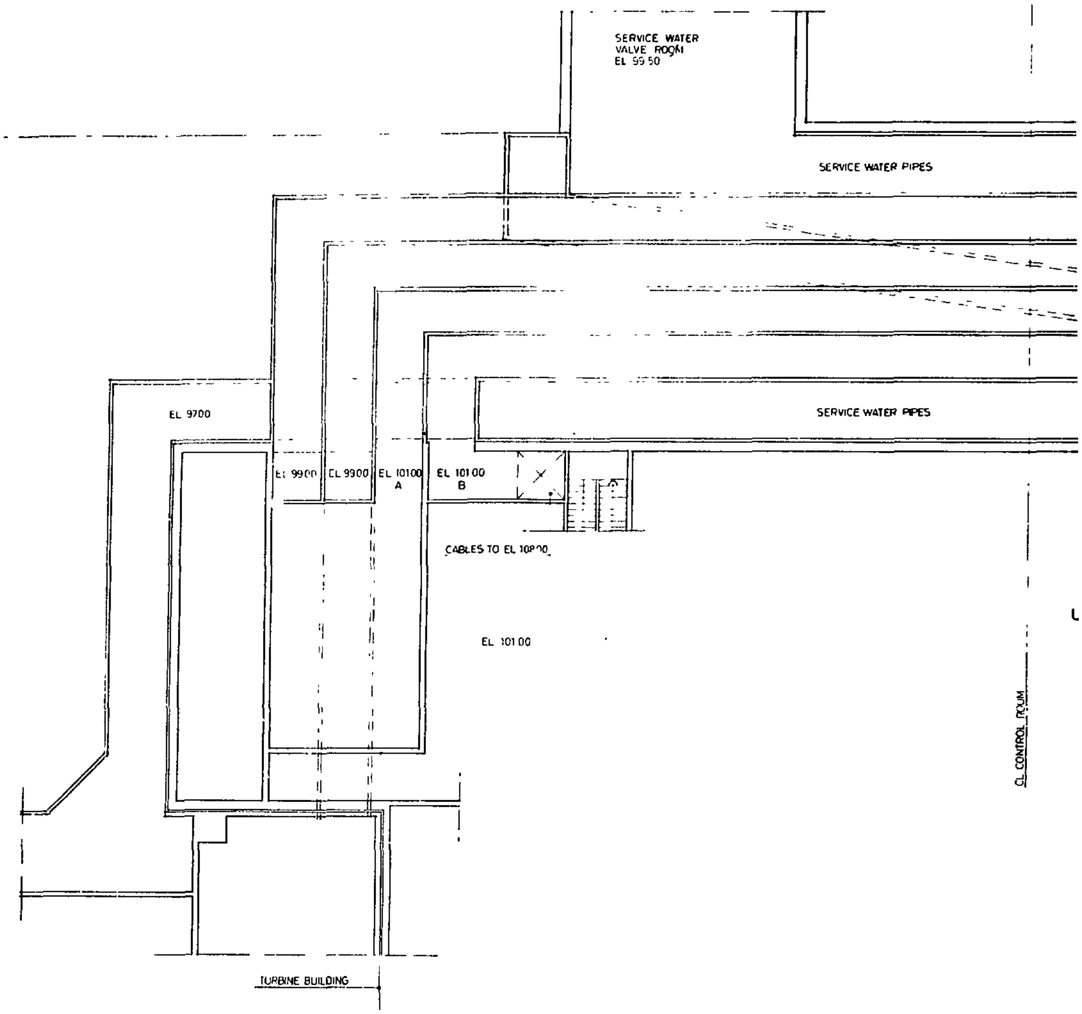
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VALVE ROOM
EL 99.50
SERVICE WATER PIPES
EL 9700
SERVICE WATER PIPES
EL 9900
CL 9900
EL 10100
A
EL 101.00
B
CABLES TO EL 108.00
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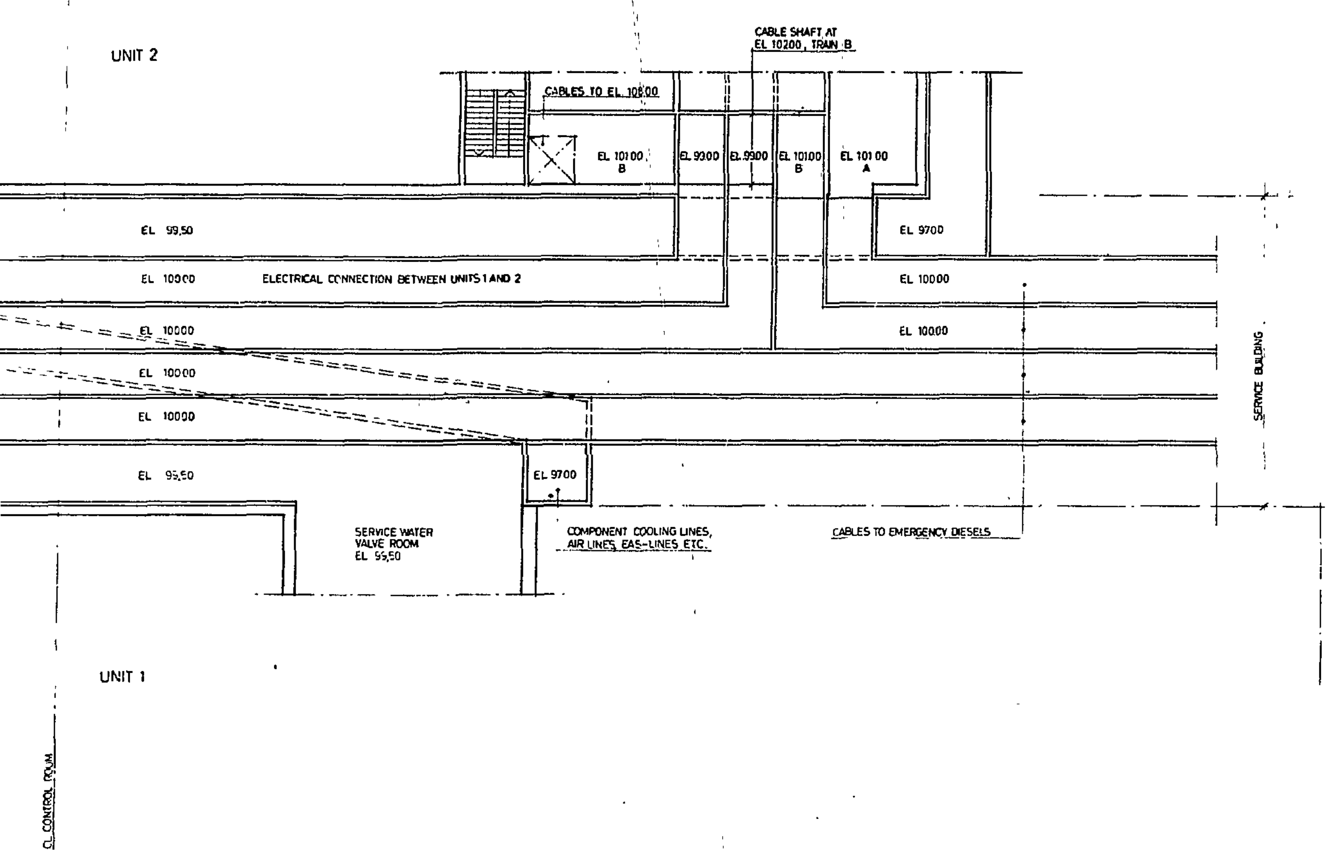
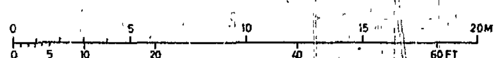
CL CONTROL ROOM
TURBINE BUILDING





WESTINGHOUSE
PROPOSAL
ALTERNATIVE No 2

NUCLEAR POWER STATION
SERVICE BUILDING
PLAN AT EL. 100,00

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MARCH 15, 1971

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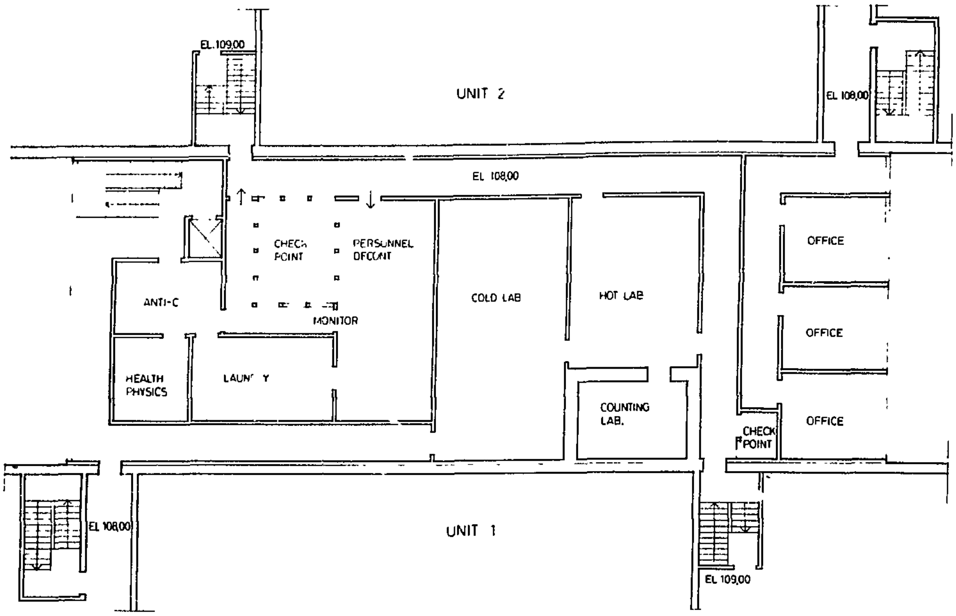

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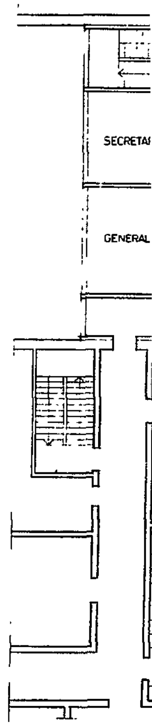

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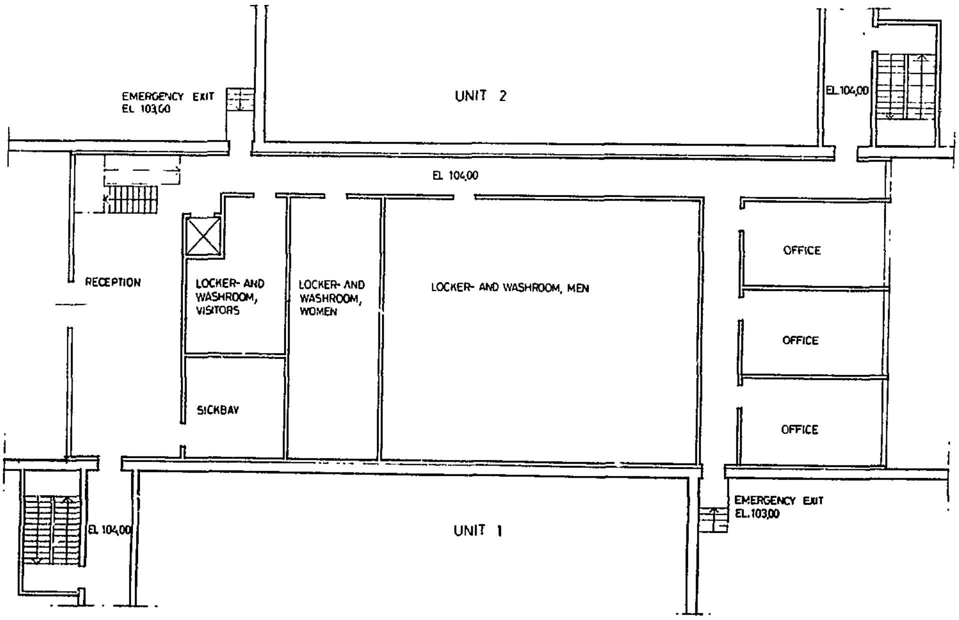

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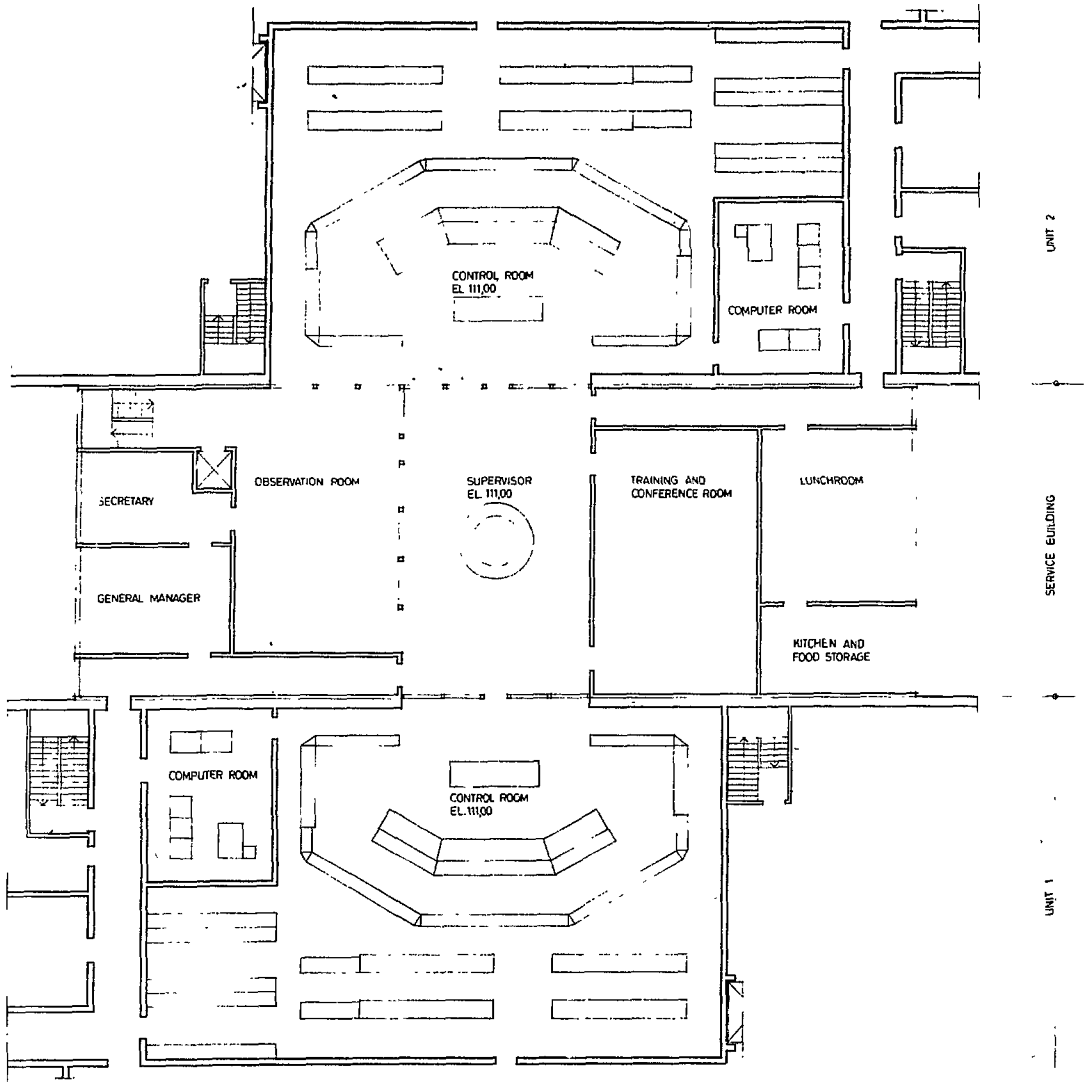

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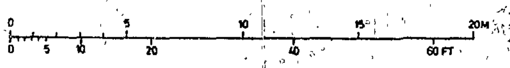

WESTINGHOUSE
PROPOSAL
ALTERNATIVE No 2

NUCLEAR POWER STATION
SERVICE BUILDING
PLAN AT EL. 104,00, 108,00 AND 111,00

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MARCH 18, 1971

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1.2.4 Nuclear Steam Supply System

The nuclear steam supply system consists of a pressurized water reactor, reactor coolant system, and associated auxiliary systems. The reactor coolant system is arranged as 3 closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump, piping, and a steam generator. An electrically heated pressurizer is connected to one of the loops.

The reactor core includes uranium dioxide pellets, enclosed in Zircaloy tubes with welded end plugs, as fuel. The tubes are supported in assemblies by structures of spring clip grids and suitable end pieces for the support of the assembled rods and restraint of abnormal axial movement. The mechanical control rods consist of clusters of stainless steel clad absorber rods which are guided by tubes located within the fuel assembly. The core consists of these fuel assemblies, loaded in 3 different enrichment regions. New fuel is introduced into the outer region, and is moved inward in a preset pattern as determined by the reactor manufacturer.

The steam generators are vertical U-tube units containing Inconel tubes. Integral separating equipment reduces the moisture content of the steam to .25 percent or less.

The reactor coolant pumps are vertical, single stage, centrifugal pumps equipped with controlled leakage shaft seals.

Nuclear auxiliary systems are provided to perform the following functions:

Accommodate reactor coolant system water requirements
Purify reactor coolant water
Introduce chemicals for corrosion inhibition
Introduce and remove chemicals for reactivity control
Cool system components
Remove residual heat during a portion of the reactor cooling period, and also when the reactor is shut down

Cool the spent fuel pool water

Permit sampling of reactor coolant water

Provide for emergency safety injection

Vent and drain the reactor coolant system and the auxiliary systems

Provide emergency containment spray

Provide containment ventilation and cooling

Dispose of liquid and gaseous wastes, and provide for disposal of solid wastes.

1.2.5 Reactor and Station Controls

The reactor is controlled by a coordinated combination of chemical shimming (boron) of the reactor coolant and mechanical control rods. The control system permits the unit to accept step load changes of ± 10 percent and ramp load changes of ± 5 percent per min over the load range from 15 percent to, but not exceeding, 100 percent power under normal operating conditions throughout most of the design lifetime subject to xenon limitations.

Control of both the reactor and turbine generator is accomplished from the main control room.

1.2.6 Waste Disposal Systems

The waste disposal systems provide all equipment necessary to collect, process, and prepare for disposal all radioactive liquid, gaseous, and solid wastes produced as a result of station operation.

Liquid wastes are collected for disposal. After sampling, some liquids are evaporated. The evaporator condensate is analyzed before discharge to the sea through the sea water discharge at concentrations below appropriate limits established by 10CFR20. x/ Evaporator residues and noncombustible solid wastes are drummed, and combustible solid wastes are baled or drummed, for shipment to an authorized off-site disposal location.

Gaseous wastes are collected and stored until radioactivity level permits discharge to the environment, after dilution, at concentrations below the established limits set forth in 10CRF20.

x/ Bl.a. på grund av den informativa karaktären har i denna koncessionsansökan de av leverantören gjorda referenserna till Code of Federal Regulations bibehållits nästan genomgående.

Det framgår av 10CFR100, fotnot 1 och 2, att de numeriska gränserna 25 och 300 rem för helkropps resp. thyreoideados inte betraktas som maximalt acceptabla dosgränser vid stora olyckor, snarare betraktas de som referensvärden för utvärderingen av reaktorns säkerhet vid den valda förläggningen.

Vid samtliga hänvisningar till 10CFR100 (liksom andra avsnitt i Code of Federal Regulations) har dessa värden trots medvetet pessimistiska antaganden kunnat underskridas med god marginal.

Beträffande 10CFR20 konstateras att dennas tillämpning ej strider mot vad som kan utläsas av ICRP.

1.2.7 Fuel Handling

The reactor is refueled with equipment designed to handle spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site. Transfer of spent fuel under water enables the use of an optically transparent radiation shield, as well as providing a reliable source of coolant for removal of residual heat.

1.2.8 Turbines and Auxiliaries

The turbine is a tandem-compound, six-flow, 3000 rpm unit. Two combined moisture separator-reheaters are employed to dry and superheat the steam between the high and low pressure stages of the unit. The turbine-generator plant can accept complete loss of external electrical load without turbine trip and without causing a reactor trip.

A single-pass, deaerating, surface condenser installed in three sections, two 100 percent design capacity steam jet air ejectors, three 50 percent design capacity condensate pumps, three 50 percent design capacity steam generator feed pumps, three auxiliary feed pumps, and five stages of feed-water heating are provided.

1.2.9 Electrical Systems

The two main generators are each a 3000 rpm, 21.5 kV, 3 phase, 50 Hz, water-inner-cooled unit. For each generator a main step-up transformer is provided to deliver power to the 400 kV switchyard.

The station service system consists of auxiliary transformers, 6.6 kV and 500 V switchgear and buses, 500 V motor control centers, 220 V AC vital bus, and 110 V and 24 V DC batteries and equipment. The normal source of station service power during plant operation is obtained from the two main generators. Power required during plant startup, shutdown and after reactor trip is supplied from 400 kV switchyard via the two main transformers.

In the event of complete loss of this normal 400 kV source, a reserve source also is available from a separate 70 kV line and switchyard via a 70/6.6 kV step-down transformer.

Four diesel engine driven generators are provided to supply power in the event of complete loss of normal and reserve station service power. Two diesel-generators have sufficient capacity for operation of all engineered safeguard equipment which must be operated in the event of a loss-of-coolant accident.

1.2.10 Engineered Safeguards Systems

The engineered safeguards systems have sufficient redundancy and independence of components and power sources, such that, under the conditions of the DBA, the systems can, even when operating with partial effectiveness, maintain the integrity of the containment and hold the exposure to the public within the criteria suggested in 10CFR100.

The systems are summarized below:

1. The steel-lined concrete containment structure provides a highly reliable barrier against the escape of radioactivity when the containment is at or above atmospheric pressure. The structure and all penetrations, including access openings and ventilation ducts, are of proven design.

2. The safety injection system provides borated water to cool the core by the injection of this water into the reactor coolant loops, by operation of the accumulators and by the safety injection pumps in major loss-of-coolant accidents.

3. The containment spray system provides sprays of borated water to the containment atmosphere. Following the DBA, the containment pressure is rapidly reduced to atmospheric pressure by the containment spray system, thereby positively terminating leakage to the atmosphere.

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1.3 GENERAL DESIGN CRITERIA

The design of the Ringhals Power Station meets the intent of the criteria as expressed within this section. Following this text of each criterion is a brief discussion specific to that criterion.

Group and title	Criterion No.
Missile Protection	40
Engineered Safety Features Performance Capacility	41
Engineered Safety Features Components Capability	42
Accident Aggravation Prevention	43
B. Emergency core cooling systems:	
Emergency Core Cooling Systems Capability	44
Inspection of Emergency Core Cooling Systems	45
Testing of Emergency Core Cooling Systems Components	46
Testing of Emergency Core Cooling Systems	47
Testing of Operational Sequence of Emergency Core Cooling Systems	48
C. Containment:	
Containment Design Basis	49
DT Requirement for Containment Material	50
Reactor Coolant Pressure Boundary Outside Containment	51
Containment Heat Removal Systems	52
Containment Isolation Valves	53
Containment Leakage Rate Testing	54
Containment Periodic Leakage Rate Testing	55
Provisions for Testing of Penetrations	56
Provisions for Testing of Isolation Valves	57
D. Containment pressure-reducing systems:	
Inspection of Containment Pressure-Reducing Systems	58
Testing of Containment Pressure-Reducing Systems	59
Testing of Containment Spray Systems	60
Testing of Operational Sequence of Containment Pressure-Reducing Systems	61

Quality Standards (Criterion 1)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

See Reply to Criterion 2

Performance Standards (Criterion 2)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) appropriated consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties standing forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Reply:

Those features of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to the mitigation of their consequences are designed, fabricated, and erected to:

(a) Quality standards that reflect the importance of the safety function to be performed. Approved design codes are used when appropriate to the nuclear application.

(b) Performance standards that enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe flooding conditions, winds, ice, or other natural phenomena characteristic of the proposed site.

Features of the facility essential to accident prevention and mitigation of accident prevention and mitigation of accident consequences are the design of the fuel, reactor coolant and containment barriers; the controls and emergency cooling systems whose function is to maintain the integrity of these three barriers; systems which depressurise the containment; power supplies and essential services to the above features; and the components employed to convey and store radioactive wastes and spent reactor fuel safety.

The fuel assembly rod design considers the effect on the Zircaloy cladding of internal fission gas pressure buildup, thermal expansion, and fabrication variations. Core design conditions and cladding material specification are selected to limit hydrogen absorption during core life to levels which will not affect fuel cladding integrity. To ensure high quality, fuel rod material is subjected to chemical analysis and tensile tests, and the rods receive dimensional inspection, X-ray of welds, ultrasonic tests, and helium leak tests.

Particular emphasis is placed on the assurance of quality of each reactor vessel to obtain material whose properties are uniformly within tolerances appropriate to the application of the design methods of the applicable code. The fatigue usage factor, derived from an assumed number of thermal cycles which is probably more than four times the number of such cycles actually expected, is less than that at which propagation of material defects would occur.

Design margin and material surveillance assure that each vessel is operated well within the ductile range of temperatures when vessel stresses are above 10,000 psi. The size of the reactor vessels is within the range of previous experience of the vessel manufacturer and of the nuclear unit designer. Further discussion of quality assurance in the reactor vessels, including the use of vessel irradiation test specimens, is given in Section 4.

Reactor containment have sufficient capacity to accept without exceeding yield stresses a combination of normal operating loads, functional loads due to hypothetical incident, and the loadings imposed by the maximum wind velocity. The emergency on-site power sources are not subject to interruption due to windstorm, floods, or to disturbances on the external power transmission system. Power cabling, motors and other equipment required for operation of the engineered safeguards are suitably protected against the effects of the hypothetical incident, and of severe external weather conditions, as applicable, to obtain a high degree of confidence in the operability of these systems should they be required.

The reference sections are as follows:

Section Title	Section
Engineered Safety Features	6
Instrumentation and Control	7
Electrical Systems	8
Auxiliary and Emergency Systems	9
Waste Disposal and Radiation Protection System	11
Förläggningsplats och omgivningar	2
Reactor	3
Reactor Coolant System	4
Containment System	5

Fire Protection (Criterion 3)

The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

Reply:
Fires or explosions occurring within the reactor facility are avoided because of inherent features in the station design.

The containment and other structures are of noncombustible construction containing mostly fire resistant equipment. Atmospheric conditions inside the containment are of a not explosive nature.

The control room is of noncombustible construction, isolated from surrounding areas by heavy concrete shielding. The control room atmosphere is not explosive and maintained under positive pressure by its air conditioning system.

The reference sections are as follows:

Section Title	Section
Reactor	3
Reactor Coolant System	4
Containment System	5
Auxiliary and Emergency Systems	9

Sharing of Systems (Criterion 4)

Reactor facilities shall not share systems or components unless it is shown that safety is not impaired by the sharing.

The reference section is as follows:

Section Title	Section
Introduction and Summary	1

Records Requirements (Criterion 5)

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.

Reply:

Records of design, fabrication and construction of essential components are maintained during the life of the reactor and are available.

Reactor Core Design (Criterion 6)

The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated.

The reactor core with its related control and protection system is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations, including the effects of the loss of reactor coolant flow, trip of the turbine-generator, loss of normal feedwater and loss of all off-site power.

Reply:

The reactor control and protection instrumentation is designed to actuate a reactor trip for any anticipated combination of station conditions when necessary to ensure a minimum departure from nucleate boiling (DNB) ratio equal to or greater than 1.30 and fuel center temperatures below the melting point of UO_2.

Reference sections are as follows:

Section Title	Section
Reactor	3
Instrumentation and Control	7
Safety Analysis	14

Suppression of Power Oscillations (Criterion 7)

The design of the reactor core with its related controls and protection systems shall ensure that power oscillations, the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed.

Reply:

The design of the reactor core and related protection systems ensures that power oscillations which could cause fuel damage in excess of acceptable limits are not possible or can be readily suppressed.

The potential for possible spatial oscillations of power distribution for this core will be reviewed as part of the core stability evaluation efforts. Part length control rods are provided to suppress these oscillations if they occur. Out-of-core instrumentation is provided to obtain necessary information concerning axial and azimuthal distributions. This instrumentation is adequate to enable the operator to monitor and control xenon induced oscillations. Based on the deviations detected by the long ion chambers, provisions in the reactor control and protection system reduce trip set-points and if necessary initiate load cut back, to maintain margin to DNB as a result of these potential oscillations in power distribution. In-core instrumentation is used to periodically calibrate and verify the information provided by the out-of-core instrumentation.

The reference sections are as follows:

Section Title	Section
Reactor	3
Instrumentation and Control	7

Overall Power Coefficient (Criterion 8)

The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

Reply:

(Statement of compliance with understanding of intent only)

The overall power coefficient is negative under normal operating conditions throughout core life.

The reference section is as follows:

Section Title	Section
Reactor	3

Reactor Coolant Pressure Boundary (Criterion 9)

The reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime.

Reply:

The Reactor Coolant System (RCS) in conjunction with its control and protective provisions is designed to accommodate the system pressures and temperatures attained under all expected modes of station operation or anticipated system interactions, and maintain the stresses within applicable code stress limits.

Fabrication of the components which constitute the pressure retaining boundary of the RCS is carried out in strict accordance with the applicable codes. In addition, there are areas where equipment specifications for RCS components go beyond the applicable codes.

The materials of construction of the pressure retaining boundary of the RCS are protected by control of coolant chemistry from corrosion phenomena which might otherwise reduce the system structural integrity during its service lifetime.

The reference section is as follows:

Section Title	Section
Reactor Coolant System	4

Containment (Criterion 10)

Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

Reply:
A prestressed concrete, steel-lined containment structure, is provided, which encloses the entire reactor coolant system. It is designed to sustain, without loss of required integrity, all effects of gross equipment failures up to and including the rupture of the largest pipe in the reactor coolant system. Engineered Safety Features, comprising Emergency Core Cooling Systems and Containment Spray Systems, then serve to cool the reactor core and return the containment to depressurized condition and maintain it at approximately atmospheric pressure for as long as the situation requires. The Containment and its associated Engineered Safety Features, therefore, meet the required functional capability of protecting the public from the consequences of gross equipment failures.

The reference sections are as follows:

Section Title	Section
Containment System	5
Engineered Safeguards	6

Control Room (Criterion 11)

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

Reply:

The control room is located at grade level in the service building. All safety related switchgear, motor generator sets, auxiliary instrument areas, battery rooms, and communications equipment are located in the basement of the service building. Emergency air-conditioning equipment is provided within the envelope of the shielded control room and associated portions of the basement. The control room is provided with the switchyard control panel, electrical recording panels, DC distribution panels, and a control panel for operation of the diesel-generator system. The control panels contain those instruments and controls necessary for operation of the station functions, such as the reactor and its auxiliary systems, turbine-generator, and the steam and power conversion systems. Selection of loading from the various station electrical distribution boards, such as the start-up boards, shutdown boards, and motor control centers, is accomplished from the station control panels.

The control room is continuously occupied by qualified operating personnel under all operating and accident conditions.

The reference sections are as follows:

Section Title	Section
Instrumentation and Control	7
Auxiliary and Emergency Systems	9

Instrumentation and Control Systems (Criterion 12)

Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables.

Reply:

Instrumentation and controls essential to avoid undue risk to the health and safety of the public, are provided to monitor and maintain neutron flux, primary coolant pressure, temperature, and control rod positions within prescribed operating ranges.

The nonnuclear regulating process and containment instrumentation measures temperatures, pressure, flow, and levels in the RCS, Steam Systems Containment and other Auxiliary Systems. Process variables required on a continuous basis for the startup, operation, and shutdown of the station are indicated, recorded, and controlled from the control room into which access is supervised. The quantity and types of process instrumentation provides ensure safe and orderly operation of all systems and processes over the full operating range of the station.

Reference sections are as follows:

Section Title	Section
Engineered Safeguards	6
Instrumentation and Control	7

Fission Process Monitors and Controls (Criterion 13)

Means shall be provided for monitoring or otherwise measuring and maintaining control over the fission process throughout core life under all conditions that can reasonably be anticipated to cause variations in reactivity of the core.

Reply:

The Nuclear Instrumentation System is provided to monitor the reactor power from source range through the intermediate range and power range up to 120 percent of full power. The system provides indication, control and alarm signals for reactor operation and protection.

The operational status of the reactor is monitored from the control room. When the reactor is subcritical, the relative reactivity status is continuously monitored and indicated by proportional counters located in instrument wells in the primary shield adjacent to the reactor vessel. Two source-detector channels are provided for supplying information on multiplication while the reactor is subcritical. A reactor trip is actuated from either channel if the neutron flux level becomes excessive.

When the reactor is critical, means for showing the relative reactivity status of the reactor is provided by control bank positions displayed in the control room. The position of the control banks is directly related to the reactivity status of the reactor when at power and any unexpected change in the position of the control banks under automatic control or change in the coolant temperature under manual control provides a direct and immediate indication of a change in the reactivity status of the reactor. Periodic sampling to determine the boric acid concentration provides a long term means of following reactivity status.

The reference section is as follows:

Section Title	Section
Instrumentation and Control	7

Core Protection Systems (Criterion 14)

Core protection systems, together with associated equipment, shall be designed to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

Reply:

The reactor protection system receives from station instrumentation, signals which are indicative of an approach to an unsafe operating condition, actuates alarms, prevents control rod motion, initiates load cutback, and/or opens the reactor trip breakers causing the insertion of the Rod Cluster Control (RCC) assemblies depending on the severity of the condition. The allowable operating range within reactor trip settings includes combinations of power, temperature and pressure which will not result in the occurrence of DNB with all reactor coolant pumps in operation.

The reference section is as follows:

Section Title	Section
Instrumentation and Control	7

Engineered Safety Features Protection Systems (Criterion 15)

Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

Reply:

Instrumentation and controls provided for the protective systems are designed to trip the reactor when necessary to prevent or limit fission product release from the core and to limit energy release; to signal closure of containment isolation valves; and to control the operation of engineered safety features equipment.

Additional tripping functions such as a high pressurizer pressure trip, low pressurizer pressure trip, high pressurizer water level trip, loss of flow trip, steam and feedwater flow mismatch trip, steam generator low-low water level trip, turbine trip, safety injection trip, nuclear source and intermediate range trips, and manual trip are provided to back up the primary tripping functions for specific accident conditions and mechanical failures.

The passive accumulators of the Emergency Core Cooling System (ECCS) do not require signal or power sources to perform their function. The actuation of the active portion of the ECCS is obtained from redundant low pressurizer pressure in coincidence with low pressurizer water level.

The Containment Isolation System provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the outside environment in the event of a loss-of-coolant accident (LOCA). The actuation of the containment isolation is by coincident and redundant containment high pressure signals, or by coincident pressurizer low level and low pressure signals.

Reference sections are as follows:

Section Title	Section
Reactor	3
Engineered Safeguards	6
Auxiliary and Emergency Systems	9

Monitoring Reactor Coolant Pressure Boundary (Criterion 16)

Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

Reply:
(Statement of compliance with understanding of intent only). Means of detecting leakage from the Reactor Coolant System is provided by measuring the airborne activity and humidity of containment and indicating changes in makeup requirements and containment sump levels.

The reference section is as follows:

Section Title	Section
Reactor Coolant System	4

Monitoring Radioactive Releases (Criterion 17)

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

Reply:

The containment atmosphere is continually monitored during normal and transient station operations, using the containment particulate and gas monitors. In the event of accident conditions, samples of the containment atmosphere are obtained via a bypass sample line arrangement to provide data of existing airborne radioactive concentrations within the containment. Radioactivity levels contained in the facility effluent discharge paths and in the environs are continually monitored during normal and accident conditions by the station radiation monitoring systems and by the radiological safety program for this facility as described in Section 11.2, "Process Radiation Monitoring", and Section 11.3, "Radiation Protection" of the Preliminary Safety Analysis Report.

The reference section is as follows:

Section Title	Section
Waste Disposal and Radiation Protection System	11

Monitoring Fuel and Waste Storage (Criterion 18)

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

Reply:

The spent fuel pool is continuously monitored for water temperature. The temperature is displayed in the control room where an audible alarm sounds, should the water temperature increase above a preset level. The radiation level above the spent fuel pool is continuously monitored by a radiation detector mounted on the fuel pit bridge. A dose rate in excess of a preset level initiates an audible and visible alarm locally and in the control room. Continuous surveillance of radiation levels in the waste storage and handling areas is maintained by an appropriately mounted radiation detector. Radiation levels in excess of preset levels initiate audio and visual alarms locally and in the control room.

The reference sections are as follows:

Section Title	Section
Auxiliary and Emergency Systems	9
Waste Disposal and Radiation Protection System	11

Protection Systems Reliability (Criterion 19)

Protection systems shall be designed for high functional reliability and in-service testability necessary to avoid undue risk to the health and safety of the public.

Reply:

The reactors use the Westinghouse magnetic-type control rod drive mechanisms (CRDM) which are similar to those used in the San Onofre, Indian Point and Connecticut Yankee stations. Upon a loss of power to the coils, the full length RCC assembly is released and falls by gravity into the core.

All reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Bypass removal of one trip circuit is accomplished by placing that circuit in a half-tripped mode; i.e., a two-out-of-three circuit becomes a one-out-of-two circuit. Testing does not trip the system unless a trip condition exists in a concurrent channel. Reliability and independence are obtained by redundancy within each tripping function.

The reference sections are as follows:

Section Title	Section
Reactor	3
Instrumentation and Control	7

Protection Systems Redundancy and Independence (Criterion 20)

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum two channels of protection for each protection function to be served.

Reply:

The Reactor Protection System is designed in accordance with the applicable criteria for standards for Nuclear Power Plant Protection Systems.

Two reactor trip breakers are provided to interrupt power to the rod drive mechanisms. The breaker main contacts are connected in series with the mechanism coils. Opening either breaker interrupts power to all mechanisms causing them to release all full length rods to fall by gravity into the core. Each breaker is opened through an undervoltage trip coil. Each protection channel actuates one reactor trip breaker undervoltage trip coil. The protection system is thus inherently safe in the event of a loss of rod control power.

The initiation of the engineered safety features provided for LOCA (e.g. ECCS pumps, and containment cooling systems) is accomplished from redundant signals derived from RCS and containment instrumentation. Channel independence is carried throughout the system from the sensors to the output relays including the power supplies for the channels.

Reference sections is as follows:

Section Title	Section
Instrumentation and Control	7

Single Failure Definition (Criterion 21)

Multiple failures resulting from a single event shall be treated as a single failure.

Reply:
Subsequent occurrences resulting from an initial failure are treated as part of the initial event, and analysis treats the entire occurrence as a single failure.

The reference section is as follows:

Section Title	Section
Safety Analysis	14

Separation of Protection and Control Instrumentation Systems (Criterion 22)

Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

Reply:

(Statement of compliance with understanding of intent only). Protection and control channels in the facility protection systems will be designed in accordance with the applicable standard Nuclear Power Plant Protection Systems.

The coincident trip philosophy will also be employed to prevent a single failure from causing a spurious trip or from defeating the function of any channel.

Each reactor trip circuit is designed so that the trip occurs upon deenergization of the circuit; and open circuit or loss of power to a channel will, therefore, result in that channel going into its trip mode. Redundancy within each channel provides reliability and independence of operation. Channel independence is carried throughout the system from the sensor to the relay providing the logic. In some cases, however, it is desirable to employ a common sensor for both a control and protection channel. Both functions are fully isolated in the remainder of the channel, control being derived from the primary safety signal path through an isolation amplifier. As such, a failure in the control circuitry does not adversely affect the safety channel.

The reference section is as follows:

Section Title	Section
Instrumentation and Control	7

Protection Against Multiple Disability for Protection Systems (Criterion 23)

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function.

Reply:

The components of the protection system are designed and arranged so that the mechanical and thermal environment accompanying any emergency situation in which the components are required to function does not interfere with that function.

The reference section is as follows:

Section Title	Section
Instrumentation and Control	7

Emergency Power for Protective Systems (Criterion 24)

In the event of loss of all off-site power, sufficient alternate sources of power shall be provided to permit the required functioning of the protective systems.

Reply:
The reactor protective instrumentation is powered from a 120 V AC vital bus which provides uninterrupted power from two 100 percent capacity, single phase, static inverters, each of which is connected to one of the station battery systems. The emergency on-site power required to operate protection systems equipment is supplied by four 50 percent capacity diesel generators.

The reference section is as follows:

Section Title	Section
Electrical Systems	8

Demonstration of Functional Operability of Protection Systems (Criterion 25)

Means shall be included for suitable testing of the active components of protection systems while the reactor is in operation to determine if failure or loss of redundancy has occurred.

Reply:

Each protection channel in service at power is capable of being calibrated and tripped independently by simulated signals for test purposes to verify its operation. This includes a checking through to the trip breakers which includes the trip logic. Thus, the operability of each trip channel can be determined conveniently and without ambiguity.

The reference section is as follows:

Section Title	Section
Instrumentation and Control	7

Protection Systems Fail-Safe Design (Criterion 26)

The protection systems shall be designed to fail into the safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electrical power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

Reply:

Each reactor trip circuit is designed so that trip occurs when the circuit is de-energized; an open circuit or loss of channel power therefore causes the system to go into its trip mode. In a two-out-of-three circuit, the three channels are equipped with separate primary sensors and each channel is energized from two independent electrical buses. Failure to de-energize when required is a mode of malfunction that affects only one channel. The trip signal furnished by the two remaining channels is unimpaired in this event.

The signal for containment isolation is developed from a two-out-of-three circuit in which each channel is separate and independent and which signals for containment isolation upon loss of power. The failure of any channel to de-energize when required does not interfere with the proper functioning of the isolation circuit.

Reactor trip is implemented by interrupting power to the magnetic latch mechanisms on each drive, allowing the full length RCC assemblies to insert by gravity. The protection system is thus inherently safe in the event of a loss of power.

The reference section is as follows:

Section Title	Section
Instrumentation and Control	7

Redundancy of Reactivity Control (Criterion 27)

Two independent control systems, preferably of different principles, shall be provided.

Reply:

One of the two reactivity control systems employs RCC assemblies to regulate the position of neutron absorber within the reactor core. The other reactivity control system employs the Chemical and Volume Control System (CVCS) to regulate the concentration of boric acid solution neutron absorber in the RCS.

Reference sections are as follows:

Section Title	Section
Reactor	3
Instrumentation and Control	7
Auxiliary and Emergency Systems	9

Reactivity Hot Shutdown Capability (Criterion 28)

The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition.

Reply:

The reactivity control systems provided are capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes. The maximum excess reactivity expected for the core occurs for the cold, clean condition at the beginning of life of the initial core.

The RCC assemblies are divided into two categories comprising a control group and shutdown groups. The control group, used in combination with chemical shim control provides control of the reactivity changes of the core throughout the life of the core at power conditions. This group of RCC assemblies is used to compensate for short term reactivity changes at power that might be produced due to variations in reactor power requirements or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life as well as those due to fuel depletion and fission product buildup.

Reference sections are as follows:

Section Title	Section
Reactor	3
Auxiliary and Emergency Systems	9

Reactivity Shutdown Capability (Criterion 29)

One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operation transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn.

Reply:

The reactor core, together with the reactor control and protection system is designed so that the minimum DNB ratio is at least 1.30 and there is no fuel melting during normal operation including anticipated transients.

The shutdown groups are provided to supplement the control group of RCC assemblies to make the reactor at least one percent subcritical (k_{eff} = 0.99) following trip from any credible operating condition to the hot, zero power condition assuming the most reactive RCC assembly remains in the fully withdrawn position. Sufficient shutdown capability will also be provided to maintain the core subcritical for the most severe anticipated cooldown transient associated with a single active failure, i.e., accidental opening of a steam bypass or relief valve. Thus shutdown capability is achieved by a combination of control rods and automatic boron addition via the ECCS with the most reactive control rod assumed to be fully withdrawn. Manually controlled boric-acid addition is used to supplement the RCC assemblies in maintaining the shutdown margin for the long term conditions of xenon decay and station cooldown.

Reactivity Holddown Capability (Criterion 30)

The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public.

Reply:
The reactivity control systems provided are capable of making and holding the core subcritical under accident conditions in a timely fashion with appropriate margins for contingencies. Normal reactivity shutdown capability by RCC assemblies is provided within two seconds following a trip signal followed by boric acid injection used to compensate for the long term xenon decay transient and for station cooldown. Any time that the reactor is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection always exceeds that quantity required for the normal cold shutdown. This quantity always exceeds the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

Reference sections are as follows:

Section Title	Section
Reactor	3
Auxiliary and Emergency Systems	9

Reactivity Control System Malfunction (Criterion 31)

The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

Reply:

Reactor shutdown with RCC assemblies is completely independent of the normal control functions since the trip breakers completely interrupt the power to the full length rod mechanisms regardless of existing control signals. The protection systems limit reactivity transients so that DNB ratio is not less than 1.3 for any single malfunction in the reactor control system or in the deboration controls.

Reference sections are as follows:

Section Title	Section
Instrumentation and Control	7
Auxiliary and Emergency Systems	9

Maximum Reactivity Worth of Control Rods (Criterion 32)

Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary, or (b) disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core.

Reply:

Limits, which include considerable margin, are placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot 1) rupture the reactor coolant pressure boundary or 2) disrupt the core, its support structures, or other vessel internals so as to lose capability to cool the core.

The wiring arrangement for the RCC assembly drive mechanisms prevents withdrawal of RCC assembly except as part of a select group of which each is part.

The maximum reactivity insertion rate is analysed in the detailed station analysis assuming two of the highest worth groups to be accidentally withdrawn at maximum speed, yielding reactivity insertion rates which are well within the capability of the overpower-overtemperature protection circuits to prevent core damage.

No credible mechanical or electrical control system malfunction can cause a rod cluster to be withdrawn at a speed greater than its mechanical limit.

Reference sections are as follows:

Section Title	Section
Reactor	3
Instrumentation and Control	7
Safety Analysis	14

Reactor Coolant Pressure Boundary Capability (Criterion 33)

The reactor coolant pressure boundary shall be capable of accomodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

Reply:
The reactor coolant boundary is shown to be capable of accommodating without rupture, the static and dynamic loads imposed as a result of a súdden reactivity insertion such as a rod ejection.

The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since control rod clusters are used to control load variations only and core depletion is followed with boron dilution, only the RCC assemblies in the controlling group are inserted in the core at power, and these rods are only partially inserted. A rod insertion limit monitor is provided as an administrative aid to the operator to ensure that this condition is met.

Reference sections are as follows:

Section Title	Section
Reactor	3
Reactor Coolant System	4
Safety Analysis	14

Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (Criterion 34)

The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure. Consideration is given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish the requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes.

Reply:

The reactor coolant pressure boundary is designed to reduce, to an acceptable level, the probability of a rapidly propagating failure.

The fast neutron exposure of the core region of the reactor vessel changes the notch toughness of the vessel material. This change is indicated by the increase in the Nil Ductility Transition Temperature (NDTT) and is allowed for in operating procedures by ensuring that the vessel is not subjected to full operating pressure until its temperature exceeds the Design Transition Temperature (DTT) defined to be NDTT plus a 60°F margin. The pressure during station startup and shutdown at temperatures below NDTT is maintained below the threshold of concern for safe operation.

The DTT dictates the procedures to be followed in the hydrostatic test and in station operations to avoid excessive cold stress. The value of the DTT is increased during the life of the station as required by the expected shift in the NDT temperature, confirmed by the experimental data obtained from irradiated specimens of reactor vessel materialsduring the station lifetime.

Reactor Coolant Pressure Boundary Brittle Fracture Prevention (Criterion 35)

Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120°F above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60°F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

Reply:

(Statement of compliance with understanding of intent only). Sufficient testing and analysis of materials employed in reactor coolant system components will be performed to insure that the required NDT limits specified in the criterion are met. Removable test capsules will be installed in the reactor vessel and removed and tested at various times in the plant lifetime to determine the effects of operation on system materials.

The reference section is as follows:

Section Title	Section
Reactor Coolant System	4

Reactor Coolant Pressure Boundary Surveillance (Criterion 36)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided.

Reply:

The design of the reactor vessel and its arrangement in the system provides the capability for accessibility during service life to the entire internal surfaces of the vessel and certain external zones of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads. The reactor arrangement within the containment provides sufficient space for inspection of the external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete.

Monitoring of the NDTT properties of the core region plates forging, weldments and associated heat treated zones are performed in accordance with ASTM E 185 (Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors). Samples of reactor vessel plate materials are retained and catalogued in case future engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens. The fracture mechanics specimens are the Wedge Opening Loading (WOL) type specimens. The observed shifts in NDTT of the core region materials with irradiation will be used to confirm the calculated limits to startup and shutdown transients.

To define permissible operating conditions below DTT, a pressure range is established which is bounded by a lower limit for pump operation and an upper limit which satisfies reactor vessel stress criteria. To allow for thermal stresses during heatup or cooldown of the reactor vessel, an equivalent pressure limit is defined to compensate for thermal stress as a function of rate of change of coolant temperature. Since the normal operating temperature of the reactor vessel is well above the maximum expected DTT, brittle fracture during normal operation is not considered to be a credible mode of failure.

The reference section is as follows:

Section Title	Section
Reactor Coolant System	4

Engineered Safety Features Basis for Design (Criterion 37)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. Such engineered safety features shall be designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

Reply:

Engineered safety features are provided to cope with any size reactor coolant pipe break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends, and to cope with any steam or feedwater line break up to and including the main steam or feedwater headers.

Limiting the release of fission products from the reactor fuel is accomplished by the ECCS which, by cooling the core, keeps the fuel in place and substantially intact and limits the metal-water reaction to an insignificant amount.

A prestressed concrete, steel-lined containment structure, is provided, which encloses the entire reactor coolant system. It is designed to sustain, without loss of required integrity, all effects of gross equipment failures up to and including the rupture of the largest pipe in the reactor coolant system. Refer to Criterion 10.

Reference section is as follows:

Section Title	Section
Engineered Safeguards	6

Reliability and Testability of Engineered Safety Features (Criterion 38)

All engineered safety features shall be designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public.

Reply:
A comprehensive program of testing will be formulated for all equipment systems and system control vital to the functioning of engineered safety features. The program consists of performance tests of individual pieces of equipment in the manufacturer's shop, integrated tests of the system as a whole, and periodic test of the activation circuitry and mechanical components to assure reliable performance, upon demand, throughout the station lifetime.

Routine periodic testing of the engineered safety features components is intended. In the event that one of the components should require maintenance as a result of failure to perform during the test according to prescribed limits, the necessary corrections or minor maintenance are made and the component is retested immediately.

Reference section is as follows:

Section Title	Section
Engineered Safeguards	6

Emergency Power for Engineered Safety Features (Criterion 39)

Alternative power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum the on-site power system and the off-site power system shall each, independently, provide this capacity, assuming a failure of a single active component in each power system.

Reply:

Four independent sections of emergency 6.6 kV bus and switchgear are provided. Each section is sized to carry at least 50 percent of the emergency load and is energized by both on-site and off-site power supplies. The on-site and off-site power supplies are both independently capable of supplying power to the Engineered Safety Features. This capability can be maintained even with the failure of any single active component in either system. In the unlikely event of total loss of station auxiliary power, the emergency 6.6 kV buses are isolated from the normal station supply. The emergency buses are energized by the four emergency diesel generators. Each diesel generator is connected to one of the emergency buses, and each bus is connected to each set of the engineered safeguards equipment, this ensuring operating of safe-guards equipment under all conditions, including a failure of a single component in each power system.

Periodic starting and loading of two emergency generators, and its emergency bus, ensures operability of the emergency generator and the automatic sequence of activating the emergency power supply in the event of loss of electrical power on site.

The reference section is as follows:

Section Title	Section
Electrical Systems	8

Missile Protection (Criterion 40)

Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

Reply:

Layout and structural design specifically protects the injection lines leading to unbroken reactor coolant loops against damage as a result of the maximum reactor coolant system pipe rupture. Separation of individual injection lines is provided to the maximum extent practicable. Movement of injection lines associated with the rupture of a reactor coolant loop is accomodated by line flexibility and by design of the pipe supports, such that no damage beyond the missile barrier is credible.

The reference sections are as follows:

Section Title	Section
Reactor Coolant System	4
Containment System	5
Engineered Safeguards	6

Engineered Safety Features Performance Capability (Criterion 41)

Engineered safety features, such as the emergency core cooling system and the containment heat removal system, shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public.

Reply:

The overall capability of the engineered safety features meets the requirements of 10CFR100 for the occurrence of any rupture of a reactor coolant or steam system pipe and including the double-ended rupture of a reactor coolant pipe, known as the Design Basis Accident (DBA).

Reference sections are as follows:

Section Title	Section
Containment System	5
Engineered Safeguards	6
Safety Analysis	14

Engineered Safety Features Components Capability (Criterion 42)

Engineered safety features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a loss-of-coolant accident to the extent of causing undue risk to the health and safety of the public.

Reply:

Instrumentation, motors, cables and penetrations located inside the containment are selected to meet the most adverse accident conditions to which they may be subjected. These items are either protected from containment accident conditions or are designed to withstand, without failure, exposure to the worst combination of temperature, pressure and humidity expected during the required operational period.

The ECCS pipes serving each loop are anchored at the missile barrier in each loop area to restrict potential accident damage to the portion of piping beyond this point. The anchorage is designed to withstand, without failure, the thrust force any branch line severed from the reactor coolant pipe and discharging fluid to the atmosphere, and to withstand a bending moment equivalent to that which produces failure of the piping under the action of free end discharge to atmosphere or motion of the broken reactor coolant pipe to which the emergency core cooling pipes are connected. This prevents possible failure at any point upstream from the support point including the branch line connection into the piping header.

Reference sections are as follows:

Section Title	Section
Containment System	5
Engineered Safeguards	6

Accident Aggravation Prevention (Criterion 43)

Protection against any action of the engineered safety features which would accentuate significantly the adverse after-effects of a loss of normal cooling shall be provided.

Reply:

The reactor is maintained subcritical following a pipe rupture accident. Introduction of borated cooling water into the core does not result in a net positive reactivity addition. The control rods insert and remain inserted.

The supply of water by the ECCS to cool hot core cladding does not produce significant water-metal reactions.

The delivery of cold emergency core cooling water to the reactor vessel following accidential expulsion of reactor coolant does not cause further loss of integrity of the RCS boundary.

The reference section is as follows:

Section Title	Section
Engineered Safeguards	6

Emergency Core Cooling System Capability (Criterion 44)

An Emergency Core Cooling System with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty.

Reply:

The ECCS employs a passive system of accumulators which do not require any external signals or source of power for their operation to cope with the short term cooling requirements of large reactor coolant pipe breaks. Two independent pumping systems, each capable of the required emergency cooling, are also provided for small break protection and to keep the core submerged after the accumulators have discharged following a large break. These systems are arranged so that the single failure of any active component does not interfere with meeting the short term cooling requirements.

Two independent pumping systems are provided, each capable of fulfilling long term cooling requirements. The failure of any single active component or the development of excessive leakage during the long term cooling period does not interfere with the ability to meet necessary long term cooling objectives with one of the systems.

The primary function of the ECCS is to deliver cooling water to the reactor core in the event of a LOCA. This limits the fuel clad temperature and thereby ensures that the core will remain intact and in place, with its essential heat transfer geometry preserved. This protection is afforded for:

a) All pipe break sizes up to and including the hypothetical circumferential rupture of a reactor coolant loop.

b) A loss of coolant associated with a rod ejection accident.

The basic design criteria for loss of coolant accident evaluations are:

1. The cladding temperature is to be less than:
   a. The melting temperature of Zircaloy-4.
   b. The temperature at which gross core geometry distortion, including clad fragmentation may be expected.

2. The total core metal-water reaction will be limited to less than 1 percent.

These criteria will assure that the core geometry remains in place and substantially intact to such an extent that effective cooling of the core is not impaired.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, ECCS adds shutdown reactivity so that with a stuck rod, no off-site power, and minimum engineered safety features, there is no consequential damage to the primary system and the core remains in place and intact. With no stuck rod, off-site power and all equipment operating at design capacity, there is insignificant (if any) cladding rupture.

Reference sections are as follows:

Section Title	Section
Engineered Safeguards	6
Safety Analysis	14

Inspection of Emergency Core Cooling System (Criterion 45)

Design provisions shall, where practical, be made to facilitate physical inspection of all critical parts of the Emergency Core Cooling System, including reactor vessel internals and water injection nozzles.

Reply:

Design provisions are made for inspection to the extent practical of all components of the ECCS. An inspection is performed periodically to demonstrate system readiness.

The pressure containment systems are inspected for leaks from pump seals, valve packing, flanged joints and safety valves during system testing.

In addition, the critical parts of the reactor vessel internals, injection nozzles, pipes, valves and safety injection pumps are inspected visually or by boroscopic examination for erosion, corrosion, and vibration wear evidence, and for nondestructive test inspection where such techniques are desirable, practical and appropriate.

The reference section is as follows:

Section Title	Section
Engineered Safeguards	6

Testing of Emergency Core Cooling System Components (Criterion 46)

Design provisions shall be made so that components of the Emergency Core Cooling System can be tested periodically for operability and functional performance.

Reply:

The design provides for periodic testing of active components of the ECCS for operability and functional performance.

Pre-operational performance tests of the components are performed in the manufacturer's shop. An initial system flow test demonstrates proper functioning of the system. Thereafter, periodic test demonstrate that components are functioning properly.

Each active component of the ECCS may be individually actuated on the normal power source at any time during station operation to demonstrate operability. The test of the safety injection pumps which perform as charging pumps during normal operation employs the minimum flow recirculation test line which connects back to the volume control tank. Remote operated valves are exercised and actuation circuits tested. The automatic actuation circuitry, valves and pump breakers also may be checked during integrated system tests performed during a planned cooldown of the RCS.

Reference sections are as follows:

Section Title	Section
Engineered Safeguards	6
Instrumentation and Control	7

Testing of Emergency Core Cooling System (Criterion 47)

Capability shall be provided to test periodically the operability of the Emergency Core Cooling System up to a location as close to the core as is practical.

Reply:

Design provisions include special instrumentation, testing and sampling lines to perform the tests during station shutdown to demonstrate proper automatic operation of the ECCS. A test signal is applied to initiate automatic action and to verify that the safety injection pumps attain required discharge heads. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.

Reference sections are as follows:

Section Title	Section
Engineered Safeguards	6
Instrumentation and Control	7

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Testing of Operational Sequence of Emergency Core Cooling System (Criterion 48)

Capability shall be provided to test, under conditions as close as practical to design, the full operational sequence that would bring the Emergency Core Cooling System into action, including the transfer to alternate power sources.

Reply:

The design provides for capability to test initially, to the extent practical, the full operational sequence up to the design conditions for the ECCS to demonstrate the state of readiness and capability of the system. This functional test is performed with the water level below the safety injection signal set-point in the pressurizer and with the RCS initially cold and at low pressure. The ECCS valving is set to initially simulate the system alignment for power operation.

The functioning of the accumulators is checked by closing the stop valve, raising the pressure in the tank and then opening the stop valve and observing the rising pressurizer level. The rising water level in the pressurizer provides indication of system delivery.

The reference section is as follows:

Section Title	Section
Engineered Safeguards	6

Containment Design Basis (Criterion 49)

The containment structure, including access openings and penetrations and any necessary containment heat removal systems, shall be designed so that the containment structure can accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a loss- of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

Reply:

The design of the containment structure is based on the Design Basis Accident (Section 14.5.2) which assumes a rupture of the largest pipe in the reactor coolant system, coupled with partial loss of the redundant engineered safeguards system (Minimum Safeguards). The maximum pressure reached for DBA is 58 psi gage (Section 14.5.5). The containment analyses performed included a 2 percent metal water reaction which is well above the less than 1 percent expected for all accidents considered. The design pressure for the containment is 60 psig.

The Containment Spray System will be designed to reduce the containment pressure following a loss of coolant accident to one half of the containment design pressure within one hour after the accident, not to exceed one half of the design pressure any time after one hour, and the containment pressure will ultimately be reduced to assentially atmospheric within 48 hours.

The reference sections are as follows:

Section Title	Section
Containment System	5
Engineered Safeguards	6
Safety Analysis	14

NDT Requirement for Containment Material (Criterion 50)

Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30°F above nil ductility transition (NDT) temperature.

Reply:

The containment liner has sufficient ductility to tolerate local deformations without rupture. The liner material has a nil ductility transition temperature of -4°F which is 64°F below the normal minimum shutdown temperature of 60°F.

The reference section is as follows:

Section Title	Section
Containment System	5

Reactor Coolant Pressure Boundary Outside Containment
(Criterion 51)

If part of the reactor coolant pressure boundary is outside the containment appropriate features, as necessary, shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features, such as isolation valves and additional containment, includes consideration of the environmental and population conditions surrounding the site.

Reply:

No portions of the reactor coolant pressure boundary extend beyond the containment barrier in that sense that all pipes penetrating the containment and directly connected to primary reactor coolant are small (< 4 inches) and protected by redundant valving as of criterion 53. All such pipes with the exception of the charging lines are also operating at reduced pressures outside the containment.

The reference section is as follows:

Section Title	Section
Containment System	5

Containment Heat Removal Systems (Criterion 52)

Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

Reply:

Two separate containment spray systems, each with 100 percent capacity and redundancy, serve to remove heat from the containment after a loss-of-coolant accident (Section 6.4). Each system contains two spray pumps of the horizontal centrifugal canned pit type. These spray pumps are located in the auxiliary building and are accessible for servicing at all times.

The reference sections are as follows:

Section Title	Section
Containment System	5
Engineered Safeguards	6

Containment Isolation Valves (Criterion 53)

Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

Reply:

All penetrations requiring valve closure for containment isolation have redundant valving so that failure of one valve does not prevent isolation of the containment. No manual operation or action will be required to activate the valves to effect isolation. All remotely actuated valves have their positions indicated in the main control room at group visual position indications.

The reference section is as follows:

Section Title	Section
Containment System	5

Containment Leakage Rate Testing (Criterion 54)

Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations, and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.

Reply:

The completed containment structure, with all necessary penetrations, is designed so that leakage does not exceed a value required to uphold stipulated dose rates for the public after DBA. The order of the leakage value may be 0.1 per cent of the contained volume per day at the design pressure of 60 psi gage. Upon completion of the containment structure and installation of all penetrations, a test of the containment is performed at 60 psi gage.

The periodic leakage rate retest is conducted at a suitable single test pressure.

Containment Periodic Leakage Rate Testing (Criterion 55)

The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

The reference section is as follows:

Section Title	Section
Containment System	5

Provision for Testing of Penetrations (Criterion 56)

Provision shall be made for testing penetrations which have resilient seals or expansion bellows to permit leaktightness to be demonstrated at design pressure at any time.

Reply:

All penetrations having resilient seals or expansion bellows are fitted with test connections to permit pressurization to 60 psi gage to demonstrate leaktightness.

The reference section is as follows:

Section Title	Section
Containment System	5

Provision for Testing of Isolation Valves (Criterion 57)

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

Reply:

Tests are performed on the isolation valves to verify their sealing capability and leaktightness (Section 5.6). Tests include valve closure and leakage tests. Isolation valves, which are normally closed, are exercised to verify closure and sealing capabilities.

The reference section is as follows:

Section Title	Section
Containment System	5

Inspection of Containment Pressure-Reducing Systems (Criterion 58)

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as pumps, valves, spray nozzles, torus, and sumps.

Reply:

Equipment comprising the engineered safeguards systems is so situated that periodic physical inspections can be made. All equipment can be inspected during planned refueling shutdowns.

The reference sections are as follows:

Section Title	Section
Containment System	5
Engineered Safeguards	6

Testing of Containment Pressure-Reducing Systems Components (Criterion 59)

The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

Reply:

The containment spray system pumps and valves can be tested, periodically, by manually tripping the required breakers in the control room to test actuation and component operation. Bypass lines on the spray pumps located outside the containment, permit flow measurements to be made, which can then be compared to preoperational tests.

Bypass lines to the refueling water storage tank permit brief operational tests of the containment spray system pumps.

The reference section is as follows:

Section Title	Section
Engineered Safeguards	6

Testing of Containment Spray Systems (Criterion 60)

A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.

Reply:

Provision is made to permit testing the systems throughout the life of the plant to assure that they are operational.

In meeting the requirements of Criterion 59, operability of the containment spray system is ensured at any time.

The reference section is as follows:

Section Title	Section
Engineered Safeguards	6

Testing of Operational Sequence of Containment Pressure-Reducing Systems (Criterion 61)

A capability shall be provided to test, under conditions as close to the design as practical, the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

Reply:

The design of the control system for the containment spray systems includes manual test switches providing for the individual testing of all the equipment in the systems and for testing of the operational sequence of the containment spray systems.

The reference sections are as follows:

Section Title	Section
Containment System	5
Engineered Safeguards	6
Instrumentation and Control	7

Inspection of Air Cleanup Systems (Criterion 62)

Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems, such as ducts, filters, fans, and dampers.

Testing of Air Cleanup Systems Components (Criterion 63)

Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

Testing of Air Cleanup Systems (Criterion 64)

A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

Testing of Operational Sequence of Air Cleanup Systems (Criterion 65)

A capability shall be provided to test, under conditions as close to design as practical, the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

Reply:

Since the containment ventilation system is not an engineered safeguard, and since it will normally be operating, no special tests or inspections are proposed for this system.

Prevention of Fuel Storage Criticality (Criterion 66)

Criticality in the new and spent fuel storage pits shall be prevented by rigid physical array, safe geometry storage racks and by fuel transfer equipment designed to handle one fuel assembly at a time.

Reply:

During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration is maintained at not less than that required to shut down the core to a $k_{eff} = 0.90$ with all RCC assemblies inserted. This concentration is sufficient to ensure that $k_{eff} \leq 0.99$, even if all RCC assemblies are withdrawn.

The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. Borated water is used to fill the spent fuel storage pool at a concentration to match that used in the reactor cavity and refueling canal during refueling operations. The fuel is stored vertically in an array with sufficient center-to-center distance between assemblies to assure $k_{eff} \leq 0.90$ even if unborated water is used to fill the pool.

The reference section is as follows:

Section Title	Section
Auxiliary and Emergency Systems	9

Fuel and Waste Storage Decay Heat (Criterion 67)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

Reply:

Decay heat from spent fuel is dissipated in the water of the fuel storage pool, and subsequently removed by a cooling system.

The reference section is as follows:

Section Title	Section
Auxiliary and Emergency Systems	9

<u>Fuel and Waste Storage Radiation Shielding (Criterion 68)</u>

Shielding for radiation protection shall be provided in the design of fuel and waste storage facilities as required to meet the requirements of 10CFR20.

Reply:

The spent fuel storage pool is designed to meet 10CFR20 requirements in providing radiation shielding for operating personnel during fuel transfer, and during storage of more than one core load of spent fuel. Work areas adjacent to the canal wall are shielded for personnel access during actual fuel transfers.

Waste storage and processing facilities in the primary auxiliary building area have shielding meeting 10CFR20 requirements for operating personnel.

Periodic surveys by health physics personnel using portable radiation detectors ensure that radiation design levels are not degraded during station lifetime.

The reference sections are as follows:

Section Title	Section
Auxiliary and Emergency Systems	9
Waste Disposal and Radiation Protection System	11

Protection against Radioactivity Release from Spent Fuel and Waste Storage (Criterion 69)

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

Reply:

Spent fuel systems are designed to preclude gross mechanical failures which could lead to significant radioactivity releases. Waste storage facilities are designed with backup containment where undue radioactivity releases might result from tank ruptures. Floor and trench drain systems provide further backup by collecting leakage which does occur.

Radioactive gases, which may result from spent fuel or waste storage tank contents release are collected by the fuel building ventilation system and the vent and drain system respectively. All discharges from those systems are monitored for particulate and gas radioactivity.

The reference sections are as follows:

Section Title	Section
Auxiliary and Emergency Systems	9
Waste Disposal and Radiation Protection System	11
Safety Analysis	14

Control of Releases of Radioactivity to the Environment (Criterion 70)

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavourable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10CFR20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10CFR100 dosage level guidelines for potential reactor reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

Reply:

Control of waste gas effluents is accomplished by holdup of waste gases in decay tanks until continual sampling of tank contents and existing environmental conditions permit discharges well within 10CFR20 requirements. In addition, waste gas effluents are monitored at the point of discharge for radioactivity and rate of flow. No decay tank burst results in an activity release greater than 25 percent of 10CFR100 limits.

Control of liquid waste effluents is maintained by batch processing of all liquids, controlled rate of release in cooling water channel, and by prevention of inadvertent tank discharge. Liquid effluents are redundently monitored for radioactivity and volyme. Liquid waste disposal system tankage and evaporator capacity are sufficient to handle any expected transient in the development of liquid waste volume.

Station solid wastes are prepared batchwise for off-site disposal by approved contractors. Solid wastes are prepared for shipment by placement in shielded and reinforced containers which meet all 10CFR20, 49CFR71-79, and AEC handbook requirements.

The reference sections are as follows:

Section Title	Section
Containment System	5
Auxiliary and Emergency Systems	9
Waste Disposal and Radiation Protection System	11
Safety Analysis	14

1.4 SAFETY CLASSIFICATION

The purpose of this section is to supplement the ANS document "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants" and to classify components in the Nuclear Steam Supply System (NSSS), Steam Systems and related components in concert with the intent of ANS criteria.

The Safety Classes, as defined in the ANS criteria and as stated below, are applied with respect to the design materials, manufacture or fabrication and operation of components and systems.

The information contained in this section reflects Westinghouse Electric interpretation of the fluid and steam system requirements which are necessary to meet the current criteria contained in the ANS document "Nuclear Safety Criteria for the Design of Stationary Pressurizer Water Plants". In addition, it represents an extension of the intent of the ANS criteria to electrical and nonfluid system mechanical components in anticipation of additional classification to be proposed by the ANS.

1.4.1 SAFETY CLASS DEFINITIONS

The safety classes applying to fluid system components as stated in the August 1970 draft issue of the ANS document, "Nuclear Safety Criteria for the design of Stationary Pressurized Water Reactor Plants" are as follows:

Safety Class 1

Safety Class 1 applies to reactor coolant system components whose failure during normal reactor operations would prevent orderly reactor shutdown and cooldown assuming makeup is provided by normal makeup systems [(1)] only. (This definition assumed ASME will include appropriate code exemptions for small pipe, fittings, and valves; e.g. instrument and sample lines).

NOTE: Boundaries of the reactor coolant system are typically defined as:

a) The reactor vessel including control rod drive mechanism housings

b) The reactor coolant side of the steam generators used to transfer reactor coolant system heat to the secondary system

c) Reactor coolant pumps

d) A pressurizer including heating and cooling provisions

e) Relief piping up to and including relief and safety valves (piping and associated tanks to receive discharges are considered outside the reactor coolant system)

f) The piping, valves and fittings needed between the principal components listed above in order to provide appropriate interconnections and flow control

g) Portions of the piping, fittings and valves leading to connecting systems up to and including the second valve (from the high pressure side) on each line provided that, if either is normally open, it must be capable of automatic closure [(2)]. If the second of two normally open check valves is considered the system boundary, then means shall be provided to periodically assess back flow

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[(1)] Normal makeup systems are those systems normally used to maintain reactor coolant inventory under respective conditions of startup, hot standby, operation or cooldown.

[(2)] Valve closure time must be such that for any postulated component failure outside the system boundary, the loss of reactor coolant would not prevent orderly reactor shutdown and cooldown assuming makeup is only be normal makeup systems.

leakage of the fist valve when closed. For a check valve to qualify as the system boundary, it must be located inside the containment system.

Safety Class 2a

Safety Class 2a applies to:

1. The containment including those valves and components of closed systems used to effect isolation of the containment atmosphere from the outside environment
2. Components of the reactor coolant system not covered in Safety Class 1
3. Safety system components of the following:

   a) Residual heat removal system
   b) Those portions of the reactor coolant auxiliary systems that form a reactor coolant letdown and makeup loop
   c) Portions of the containment cooling (spray) systems that may recirculate reactor coolant
   d) Portions of the emergency core cooling system that may recirculate reactor coolant
   e) Portions of the steam and normal feedwater systems extending from and including the secondary side of the steam generator up to and including the outermost containment isolation valves.[1]

Safety Class 2b

Safety Class 2b applies to safety system components of the following:

   a) Those portions of the reactor auxiliary systems that provide boric acid for the letdown and makeup loop.
   b) The containment cooling and air cleanup systems excluding these portions of the spray subsystem covered in Class 2a.

(1) Isolation valves may include steam line stop valves, check valves, pressure relief valves and safety valves.

c) Accumulator and refueling water supply subsystem of the emergency core cooling system
d) Emergency feedwater system
e) Portions of component and process cooling systems that cool other safety systems

Safety Class 3

Safety Class 3 applies to components, the failure of which would result in release to the environment of gaseous radioactivity normally heldup of the following:

a) Portions of the reactor coolant auxiliary systems that service the letdown and makeup loop
b) Portions of the waste disposal system

1.4.2 CODE AND CODE CLASSES

The Code and Code Class relative to the 1971 Edition of ASME Section III and the comparison with the previous codes is as follows:

ASME III 1971 Edition Class	PREVIOUS CODE AND CODE CLASSES		
	Vessel	Pumps and Valves	Pipe and Fittings
1	ASME III Class A	ASME P&V Class I	ANSI B31.7 Class I
Metal Containment	ASME III	-	-
2	ASME III Class C	ASME P&W Class II	ANSI B31.7 Class II
3	ASME VIII	ASME P&W Class III	ANSI B31.7 Class III

1.4.3 COMPONENT CLASSIFICATION

The component classification is given in Tables 1.4.3-1 and 1.4.3-2.

SYSTEM / COMPONENT	ANS SAFETY CLASS	CODE	CODE CLASS	Q.A.	ISI (2)	REMARKS
REACTOR COOLANT SYSTEM						
Reactor vessel	1	ASME III	1	(9)	Yes	
Full Length CRDM Housing	1	ASME III	1	(9)	Yes	ISI may not be required, see ISI 120 (d) of ASME Section XI
Part Length CRDM Housing	1	ASME III	1	(9)	Yes	
Reactor Coolant Pump Assembly	1	ASME III	1	(9)	Yes	See Note 8
Reactor Coolant Pump Casing	1	ASME III	1	(9)	Yes	
Reactor Coolant Pump Internals	1	ASME III	1	(9)	Yes	
Reactor Coolant Pump Motor	2b	NEMA 20		(9)	No	In service inspection not required, however, flywheel must be inspected periodically
Steam Generator (tube side)	1	ASME III	1	(9)	Yes	
(shell side)	2a	ASME III	2	(9)	No	See Note 7a
Pressurizer	1	ASME III	1	(9)	Yes	
Reactor Coolant Piping, Fittings and Fabrication	1	ASME III	1	(9)	Yes	
Surge Pipe, Fittings & Fabrication	1	ASME III	1	(9)	Yes	
Bypass Manifold	1	ASME III	1	(10)	Yes	
RC Thermowells	1	ASME III	1	(10)	Yes	
Safety Valves	1	ASME III	1	(9)	Yes	
Relief Valves	1	ASME III	1	(9)	Yes	
Valves to RC System boundary	1	ASME III	1	(9)	Yes	
Piping to RC System boundary	1	ASME III	1	(9)	Yes	
Pressurizer Relief Tank	NNS	ASME VIII	UW	(10)	No	NNS denotes Non-Nuclear Safety
CRDM Head Adapter Plugs	1	ASME III	1	(10)	No	

(1) All notes are on last page.

COMPONENT	ANS SAFETY CLASS	CODE	CODE CLASS	Q.A.	ISI REQD	REMARKS
CHEMICAL & VOLUME CONTROL SYSTEM						
Regenerative HX	2a	ASME III	2	(9)	No	
Letdown HX (tube side)	2a	ASME III	2	(9)	No	
(shell side)	2b	ASME III	3	(9)	No	See Note 7b
Mixed bed demineralizer	3	ASME III	3	(10)	No	
Deborating demineralizer	3	ASME III	3	(10)	No	
Cation bed demineralizer	3	ASME III	3	(10)	No	
Reactor coolant filter	2a	ASME III	2	(9)	No	
Volume Control Tank	2a	ASME III	2	(9)	No	
Charging pumps centrifugal	2a	ASME III	2	(9)	No	See Note 7c and 8
Positive Displacement Pump	NNS	ASME III	3	(10)	No	Hydro Pump Only
Seal water injection filter	2a	ASME III	2	(9)	No	
Letdown orifices	2a	ASME III	2	(10)	No	
Excess letdown HX (Tube side)	2a	ASME III	2	(9)	No	
(Shell side)	2b	ASME III	3	(9)	No	See Note 7b
Seal water return filter	2a	ASME III	2	(9)	No	
Seal water HX (Tube side)	2a	ASME III	2	(9)	No	
(Shell Side)	2b	ASME III	3	(9)	No	See Note 7b
Boric acid tanks	2b	ASME III	3	(9)	No	Note 7d for 12% BA
Boric acid filter	2b	ASME III	3	(9)	No	
Boric acid transfer pump	2b	ASME III	3	(9)	No	Note 7d
Boric acid blender	2b	ASME III	3	(10)	No	
Resin fill tank	NNS	ASME VIII	UW	(10)	No	
Boric acid batching tank	2b	ASME III	3	(10)	No	

SYSTEM	COMPONENT	ANS SAFETY CLASS	CODE	CODE CLASS	Q.A.	ISI REQD	REMARKS
CHEMICAL & VOLUME CONTROL SYSTEM							
	Chemical mixing tank	NNS	ASME VIII	UW	(10)	No	
	Chemical mixing tank Orifice	NNS	No Code		(11)	No	
	Charging pump accumulator (where applicable for positive displ. pumps)	2a	ASME III	2	(10)	No	
	RCP No. 1 Seal Bypass Orifice	1	ASME III	1	(10)	No	
	System Piping						See Comment 1
	System Valves						See Comment 1

Comment 1 Classification of piping and valves shall be as defined by Systems Engineering flow diagrams for the appropriate safety class. Code, Code Class and Q.A. level shall be those required by the Safety Class.

SYSTEM / COMPONENT	ANS SAFETY CLASS	CODE	CODE CLASS	Q.A.	ISI REQD	REMARKS
BORON RECYCLE SYSTEM						
Holdup tank	3	(4)	(4)	(9)	No	
Holdup tank recirculation pumps	3	ASME III	3	(10)	No	
Gas Stripper Feed Pump	3	ASME III	3	(10)	No	
Evaporator feed ion exchanger	3	ASME III	3	(10)	No	
Ion exchanger filter	3	ASME III	3	(10)	No	
Gas stripper package (where applicable)	3	ASME III ASME III ASME III	3 (VES) 3 (P&V) 3 (Pipe)	(9) (9) (9)	No	See Note 8
Boric Acid evaporator package W/Gas Stripper	3	ASME III ASME III ASME III	3 (VES) 3 (P&V) 3 (Pipe)	(9) (9) (9)	No	See Note 8
Evaporator condensate filter	NNS	ASME VIII	UW	(10)	No	
Evaporator condensate demin.	NNS	ASME VIII	UW	(10)	No	
Monitor tanks	NNS	ASME VIII	UW	(10)	No	
Monitor tank pumps	NNS	ASME III	3	(10)	No	
Concentrates filter	NNS	ASME VIII	UW	(10)	No	
Concentrates holding tank	NNS	ASME VIII	UW	(10)	No	
Concentrates holding tank transfer pump	NNS	ASME III	3	(10)	No	
Reactor makeup water storage tank	2b(3)	(4)	(4)	(9)	No	
Reactor makeup water pumps	2b(3)	ASME III	3	(9)	No	

SYSTEM / COMPONENT	ANS SAFETY CLASS	CODE	CODE CLASS	Q.A.	ISI REQD	REMARKS
SAFETY INJECTION SYSTEM						
Refueling water storage tank	2b	(4)	(4)	(9)	No	
Accumulators	2b	ASME III	3	(9)	No	
High head SIS pumps	2a	ASME III	2	(9)	No	See Note 7c and 8
Low Head SIS Pump	2a	ASME III	2	(9)	No	See Note 7c and 8
Boron injection tank (discharge)	2a	ASME III	2	(9)	No	See Note 7d
BIT-Recirculation Pump (where applicable)	2b	ASME III	3	(9)	No	See Note 7d
Boron Injection Surge Tank (where applicable)	2b	ASME III	3	(9)	No	
S. I. System Piping	2a/2b	ASME III	2/3	(9)	No	See Comment 1
S. I. System Valves	2a/2b	ASME III	2/3	(9)	No	See Comment 1

Comment 1 Exclusive of valves and piping in RC system boundary which are Class 1. Classification of piping and valves shall be derived from the Safety Classes identified on the Engineering Flow Diagrams.

SYSTEM / COMPONENT	A-S SAFETY CLASS	CODE	CODE CLASS	Q.A.	ISI REQD	REMARKS
RESIDUAL HEAT REMOVAL LOOP						
Residual heat removal pump	2a	ASME III	2	(9)	No	See Note 7c and 8
Residual heat exchanger (Tube side)	2a	ASME III	2	(9)	No	
(Shell side)	2b	ASME III	3	(9)	No	See Note 7b
Residual heat removal piping	2a	ASME III	2	(9)	No	See Comment 1
Residual heat removal valves	2a	ASME III	2	(9)	No	See Comment 1
CONTAINMENT SPRAY SYSTEM						
Spray Additive Tank	2b	ASME III	3	(9)	No	
Containment Spray Pump	2a(5)	ASME III	2	(9)	No	See Note 7c and 8
S.I. System Eductor	2a	ASME III	2	(10)	No	

Comment 1 - Exclusive of valves and piping in R.C. pressure boundary which are Class I. Classification of piping and valves shall be derived from the Safety Class's identified on the engineering flow diagrams.

SYSTEM / COMPONENT	ANS SAFETY CLASS	CODE	CODE CLASS	Q.A.	ISI REQD	REMARKS
COMPONENT COOLING						
Component cooling heat exchanger	2b	ASME III	3	(9)	No	See Note 7c
Component cooling pumps	2b	ASME III	3	(9)	No	See Note 7d
Component cooling surge tank	2b	ASME III	3	(9)	No	
Component cooling piping	2b	ASME III	3	(9)	No	Except for valves and piping required for containment isolation which are Class 2a, Code Class 2
Component cooling valves	2b	ASME III	3	(9)	No	
SPENT FUEL PIT						
Spent fuel pit heat exchanger	NNS	ASME VIII	UW	(10)	No	See Note 8
Spent fuel pit pump	NNS	ASME III*	3	(10)	No	
Skimmer pump	NNS	No Code		(11)	No	
Spent fuel pit filter	NNS	ASME VIII	UW	(10)	No	
Spent fuel pit demineralizer	NNS	ASME VIII	UW	(10)	No	
Spent fuel pit piping	NNS	ANSI B31.1	3	(10)	No	
Spent fuel pit valves	NNS	ASME III	3	(10)	No	

COMPONENT	SAFETY CLASS	CODE	CODE CLASS	Q.A.	ISI REQD	REMARKS
WASTE DISPOSAL SYSTEM						
Reactor coolant drain tank	NNS	ASME VIII	UW	(10)	No	
Reactor coolant drain tank pump(A)	NNS	ASME III	3	(10)	No	
Reactor coolant drain tank pump(B)	NNS	ASME III	3	(10)	No	
Chemical drain tank	NNS	ASME VIII	UW	(10)	No	
Chemical drain pump	NNS	ASME III	3	(10)	No	
Laundry and hot shower tanks	NNS	ASME VIII	UW	(10)	No	
Laundry tank pump	NNS	ASME III	3	(10)	No	
Laundry tank basket strainer	NNS	ASME VIII	UW	(11)	No	
Sump tank	NNS	ASME VIII	UW	(10)	No	
Sump tank pump	NNS	ASME III	3	(10)	No	
Waste holdup tank	3		3	(9)	No	Non Nuclear Safety if vented to other than vent header
Waste evaporator feed pump	3	ASME III	3	(10)	No	
Waste evaporator package	3	ASME III	3 (VES)	(9)	No	See Note 8
		ASME III	3 (P&V)	(9)		
		ASME III	3 (Pipe)	(9)		
Spent resin storage tank	3	ASME III	3	(9)	No	
Waste condensate tank	NNS	ASME VIII	UW	(10)	No	
Waste condensate pumps	NNS	ASME III	3	(10)	No	
Waste gas compressor package	3	ASME III	3 (VES)	(9)	No	See Note 8
		ASME III	3 (P&V)	(9)		
		ASME III	3 (Pipe)	(9)		
Gas decay tanks	3	ASME III	3	(9)	No	
Waste filter	NNS	ASME VIII	UW	(10)	No	
Containment sump pumps	NNS	None		(11)	No	
Hydrogen manifold	NNS	(6)		(11)	No	
Nitrogen manifold	NNS	(6)		(11)	No	
Auxiliary building sump pumps	NNS	None		(11)	No	
Heat removal pump pit sump pumps	2a	ASME III	2	(9)	No	
Waste Disposal Valves	NNS	ASME III	3	(10)	No	
Waste Disposal Piping	NNS	ANSI B31.1		(10)	No	

1. The explanation of terms used [illegible] equipment classification list are defined in Section 1.4.1

2. Identifies if Inservice Inspection (ASME Section XI) is applicable.

3. Class 2b when used as source of [illegible] for Component Cooling System, otherwise non-nuclear safety related.

4. Storage vessel not exceeding 15 psig and 200°F to be designed, fabricated and tested to API 620, API 650 or AWWA D100.

5. Safety Class 2b, Code Class 3 if not used for recirculation.

6. National Fire Underwriters and Underwriters Laboratory Certification.

7. Necessary services required to perform the safety function:

   7a Feedwater and controlled steam bleed
   7b Component cooling water
   7c Emergency power plus component cooling water
   7d Emergency power
   7e Service water

8. Portions of equipment containing component cooling water shall be SC-2b, Code Class 3.

9. Quality Administration:

   Westinghouse QCS-#1 and Section 10 of Westinghouse Administrative Specification
   or
   ASME Appendix IX with Security Survey (Inspection check list must be submitted to Westinghouse) and Section 10 of W Administrative Specification.

10. Quality Administration:

    Westinghouse Administrative Specification Section 10 and ASME Code Inspection at Source, if applicable.

11. Quality Administration:

    Westinghouse rights to inspect, no Quality Administration required.

STEAM SYSTEMS [illegible] EQUIPMENT [illegible]LASSIFICATION

COMPONENT	ANS SAFETY CLASS	CODE	CODE CLASS	Q.A. LEVEL (6)	REMARKS
MAIN STEAM (To First Stop Valve Outside Containment					
Piping	2a	ASME III	2		
Stop Valve	2a	ASME III	2		
Steam Generator Safety Valves	2a	ASME III	2		
Steam Generator Relief Valves	2a	ASME III	2		
Pipe Branches & Connections	2a	ASME III	2		
MAIN FEED (To First Automatic Stop or Check Valve Outside Containment)					
Piping	2a	ASME III	2		
Stop & Non-return Valves	2a	ASME III	2		
Pipe Branches & Connections	2a	ASME III	2		
AUXILIARY FEED DELIVERY					
Pumps	2b	ASME III	3		
Piping	2b(4)	ASME III	3		
Valves	2b(4)	ASME III	3		
AUXILIARY FEED ASSURED SUPPLY					
Tanks	2b	(2)	(2)		
Piping	2b	ASME III	3		
Valves/Pumps	2b	ASME III	3		

STEAM SYSTEMS MECHANICAL EQUIPMENT CLASSIFICATION

Table 1.4.3-2

COMPONENT	ANS SAFETY CLASS	CODE	CODE CLASS	Q.A. LEVEL (6)	REMARKS
<u>STEAM GENERATOR BLOWDOWN</u> (That Portion Which is Part of Containment Boundary)					
Piping	2a	ASME III	2		Remainder of Systems
Valves	2a	ASME III	2		Non-Nuclear Safety
<u>STEAM GENERATOR SAMPLING</u>					
Piping	2b(4)	ASME III	3		
Valves	2b(4)	ASME III	3		
<u>SERVICE WATER (PORTIONS USED FOR SAFEGUARDS)</u>					
Piping	2b(4)	ASME III	3		
Pumps/Valves	2b(4)	ASME III	3		
Screens/Strainers	2b	(5)	(5)		
<u>AUXILIARY POWER GENERATION</u>					
Power Unit	2b	(5)	(5)		
Fuel Oil Tanks	2b	(5)	(5)		
Fuel Oil Valves/Pumps	2b	(5)	(5)		
Fuel Oil Piping	2b	(5)	(5)		

STEAM SYSTEMS M[illegible] [illegible]SIFICATION

Table 1.4.3-2

COMPONENT	NS SAFETY CLASS	CODE	CODE CLASS	Q.A. LEVEL (6)	REMARKS
ENGINE COOLING					
Valves/Pumps	2b	(5)	(5)		
Piping	2b	(5)	(5)		
Tanks	2b	(5)	(5)		
Heat Exchanger	2b	(5)	(5)		
ENGINE STARTING					
Valves/Pumps	2b	(5)	(5)		
Piping	2b	(5)	(5)		
Tanks	2b	(5)	(5)		
FIRE PROTECTION (Inside Containment and For Safety Systems)					
Pumps/Piping/Valves	2b	(3) (5)			
Hoses/Hydrants/Nozzles	2b	(3) (5)			
Portable Extinguishers	2b	(3)			

Notes

1) This listing defines the safety equipment for the Steam Systems Area defined in the ANS criteria, and is typical of all W PWR plants.

   Explanation of terms used in this equipment classification list is included in Section 1.4.1.

2) Storage vessel not exceeding 15 psig and 200°F to be designed, fabricated and tested to API 620, API 650 or AWWA D100.

3) National Fire Underwriters and Underwriters Laboratory Certification. Interaction of fire protection systems and plant equipment should be considered.

4) Safety Class 2a where part of containment boundary.

5) Accepted practices that produce equivalent quality equipment required by the safety class.

6) The purchaser must require the manufacturer, through appropriate specifications, to design and fabricate safety equipment within the framework of a controlled quality system. All equipment not classified is considered to be non-nuclear safety.

KAPITEL 2

Förläggningsplats och omgivningar

RINGHALS 3

2. FÖRLÄGGNINGSPLATS OCH OMGIVNINGAR

Det tredje aggregatet placeras c:a 300 m söder om de två första aggregaten vilka är under uppförande. Det första aggregatet planeras tas i kommersiell drift omkring årsskiftet 1973-74 och aggregat två under juli månad 1974.

Data rörande förläggningsplatsen och dess omgivningar har redovisats i ansökan om koncession enligt atomenergilagen ingiven den 29 mars 1967 och beviljad av Kungl. Maj:t den 6 juni 1968. Säkerhetsredovisningar har inlämnats till delegationen för atomenergifrågor för de båda första aggregaten. Delegationen kommer att fortlöpande delges resultat av meteorologiska och hydrologiska undersökningar på platsen.

2.1 STATIONSOMRÅDET

2.1.1. Situationsplaner

På översiktig situationsplan, bilaga 2:1, redovisas anläggningen med planerade områden för aggregaten tre och fyra, vattenvägar m m liksom platsen för de båda första aggregaten. Bilaga 2:2 visar själva stations området med de olika byggnadernas inbördes placering.

2.1.2. Stadsplan

Stadsplan har upprättats för stationsområde omfattande aggregaten ett t o m fyra. Planen fastställdes av Kungl. Maj:t den 24.1.1969.

2.1.3. Byggnadsförbud runt anläggningen

Länsstyrelsen meddelade den 14.11.1968 förbud mot viss nybyggnad inom en del av Väröhalvön. Länsstyrelsens resolution har tidigare redovisats för delegationen.

2.1.4. Vattendom

I den av Västerbygdens vattendomstol meddelade deldomen den 19.2.1969 innefattades även verkställighetstillstånd. Domen överklagades av enskilda sakägare. Vattenöverdomstolen fastställde deldomen den 16 december 1969. Vattenöverdomstolens dom överklagades och ärendet behandlas för närvarande av Högsta domstolen. Västerbygdens vattendomstol uppsköt i sin deldom prövning av skumbalkens utförande samt utsläpp av avloppsvatten för aggregat tre och fyra. En redovisning i detta hänseende är för närvarande under utarbetande.

2.2 HAMN

Vattenfall har av Västerbygdens vattendomstol erhållit tillstånd att anlägga en hamn vid Båtfjorden alldeles innanför Videbergs fiskehamn, se bilaga 2:1. Hamnen är under byggnad och har dimensionerats så att reaktortankar och annat skrymmande gods kan tas i land. Den skall även användas för utskeppning av utbrända bränsleelement m m. Hamnen har ett vattendjup av c:a 6 m.

2.3 METEOROLOGI

Klimatologiska undersökningar pågår avseende nederbörd, lufttemperatur, dimfrekvens etc. Vidare planeras lokalmeteorologiska undersökningar att utföras under minst ett år före idrifttagning av det första aggregatet.

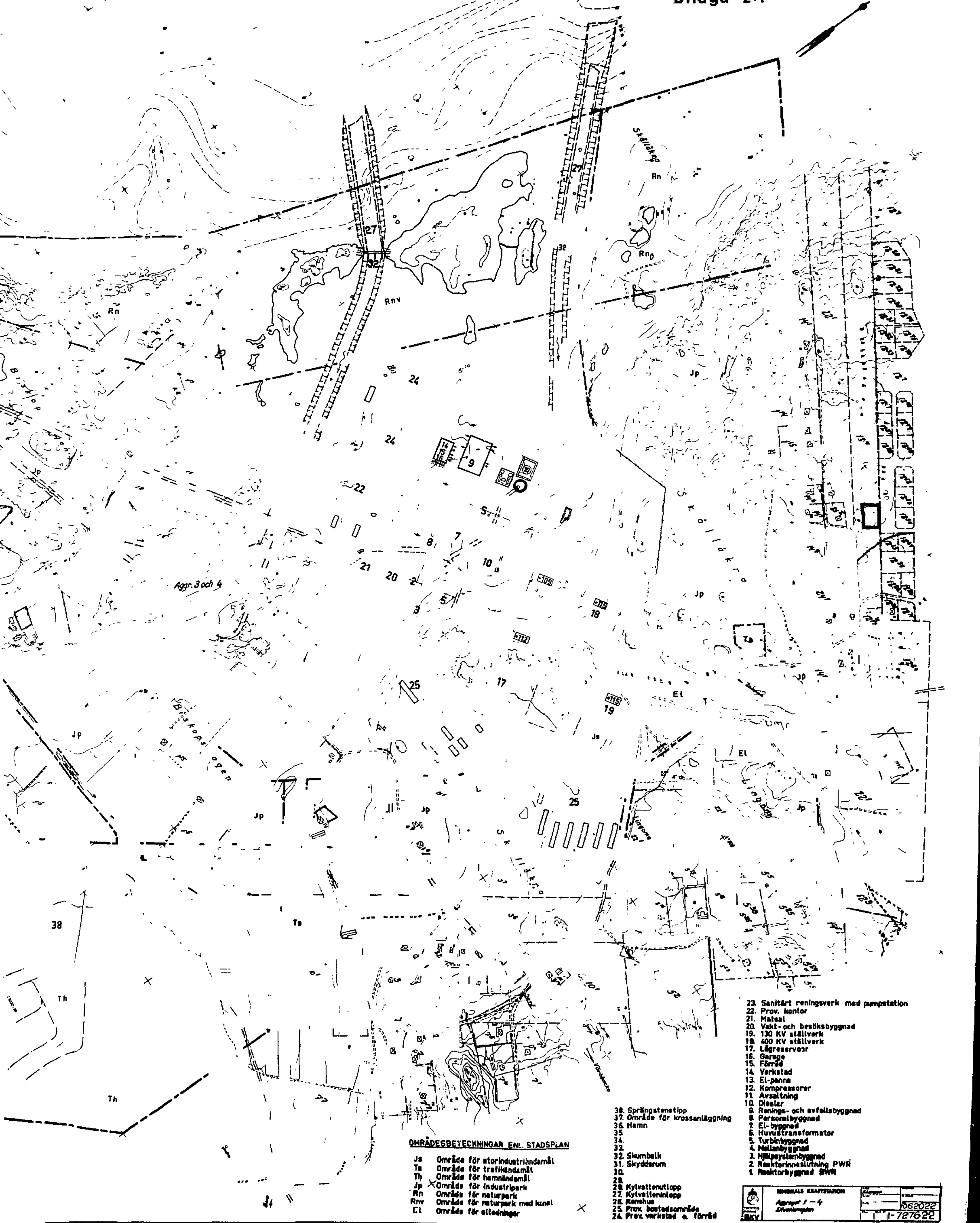
2.4 HYDROLOGI

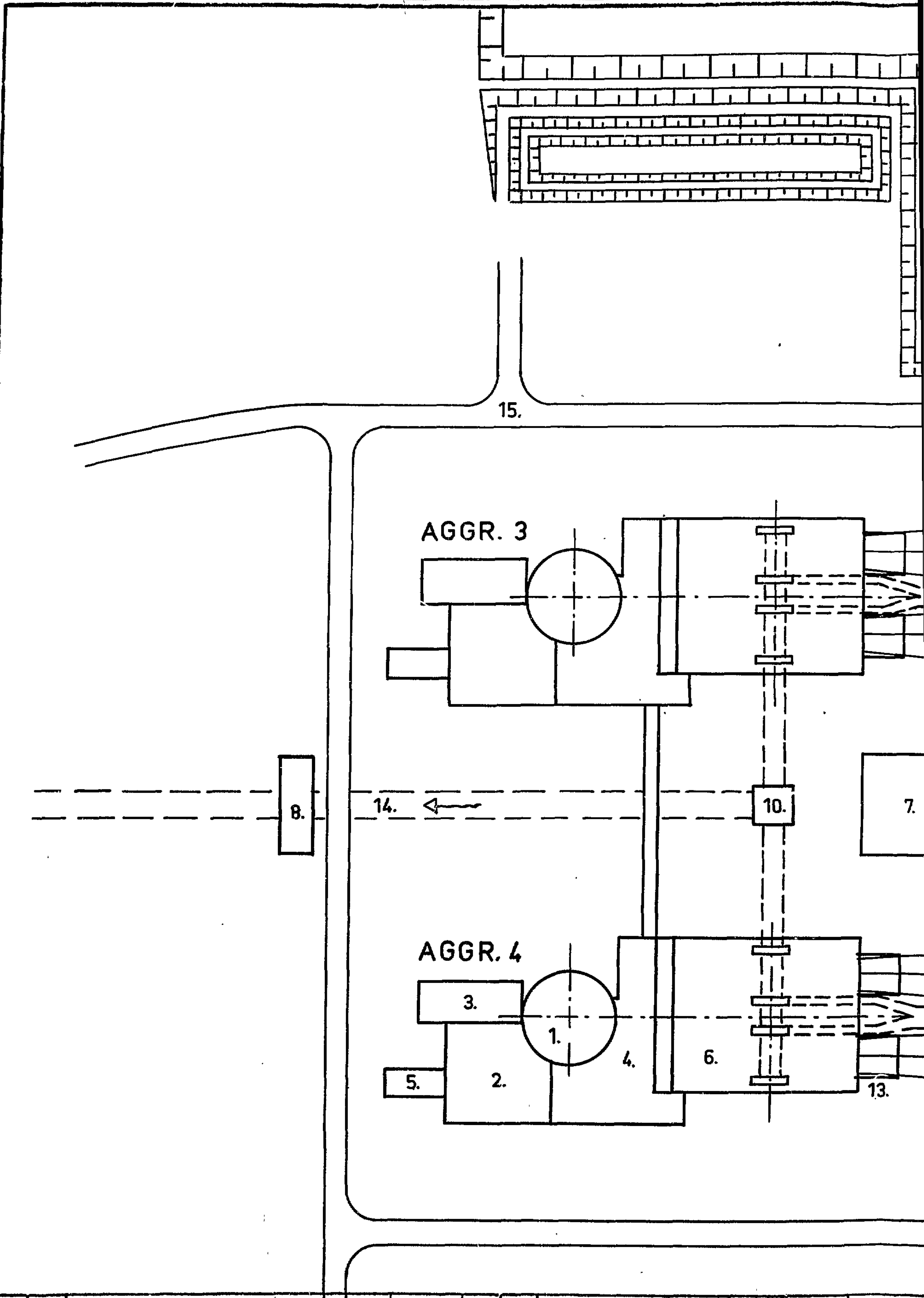
SMHI utför hydrologiska undersökningar med mätningar av vattentemperaturen, salthalt och strömningar m m. Statens Naturvårdsverk deltar även i dessa undersökningar.

2.5 FISKE

Omfattande fiskeribiologiska undersökningar utföres av Statens Naturvårdsverk för att fastställa rådande förhållanden. Undersökningarna kommer även att bedrivas sedan kraftstationen tagits i drift.

28
38
36
Th
Jp
Rn
0
100
200
300
400
500 M

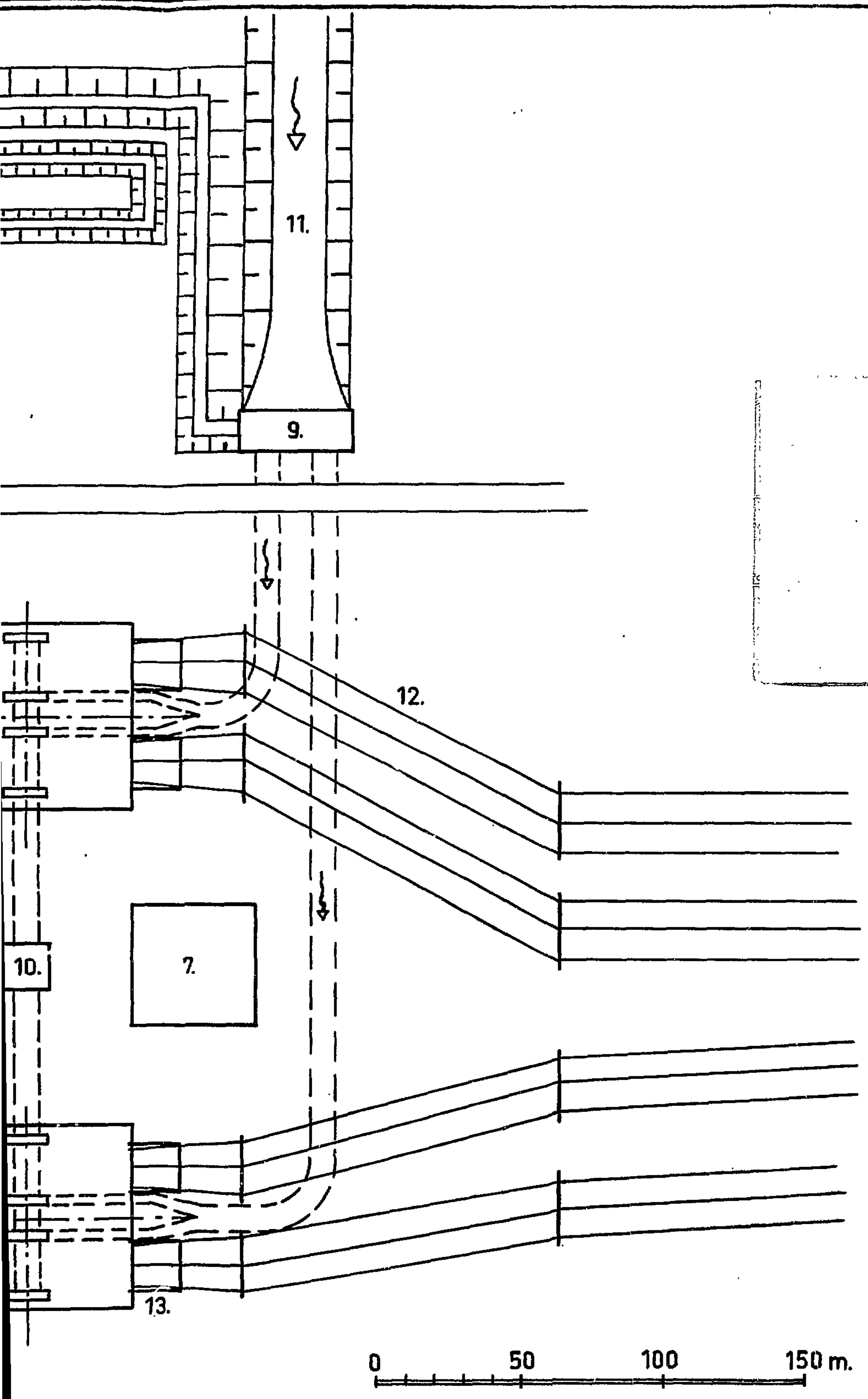

Bilaga 2:1
Skällsjö
Rn
Rnv
Rn0
Jp
Ta
Th
Js
El
T
Aggr. 3 och 4
Biskopsvägen
Skälldkra
Lingonv.
38
ОMRÅDESBETECKNINGAR ENL. STADSPLAN
Js Område för storindustriändamål
Ta Område för trafikändamål
Th Område för hamnändamål
Jp Område för industripark
Rn Område för naturpark
Rnv Område för naturpark med kanal
El Område för elledningar
38. Sprängstenstipp
37. Område för krossanläggning
36. Hamn
35.
34.
33.
32. Skumbalk
31. Skyddsrum
30.
29.
28. Kylvattenutlopp
27. Kylvatteninlopp
26. Renshus
25. Prov. bostadsområde
24. Prov. verkstad o. förråd
23. Sanitärt reningsverk med pumpstation
22. Prov. kontor
21. Matsal
20. Vakt- och besöksbyggnad
19. 130 KV ställverk
18. 400 KV ställverk
17. Lågreservoar
16. Garage
15. Förråd
14. Verkstad
13. El-panna
12. Kompressorer
11. Avsaltning
10. Diesler
9. Renings- och avfallsbyggnad
8. Personalbyggnad
7. El-byggnad
6. Huvudtransformator
5. Turbinbyggnad
4. Mellanbyggnad
3. Hjälpsystembyggnad
2. Reaktorinneslutning PWR
1. Reaktorbyggnad BWR
Aggregat 1 – 4
062022
1-727622



	—	Ursprungligt utförande	—	—	—		
Avv. åtg.	Nat	Ändring	Dat.	Inf.	Godk.	Tillhörande ritningar etc.	Ritn.

Bilaga 2:2

18.1.72



15. VÄG
14. KYLVATTENUTLOPP
13. TRANSFORMATORBÅS
12. KRAFTLEDNING
11. KYLVATTENINLOPP
10. SVALLSCHAKT
9. RENSHUS
8. KONTORSBYGGNAD
7. SERVICEBYGGNAD
6. TURBINBYGGNAD
5. AKTIV VERKSTAD
4. ELBYGGNAD
3. BRÄNSLEBYGGNAD
2. HJÄLPSYSTEMBYGGNAD
1. REAKTORBYGGNAD

VATTENFALL Avdelning EVS 4	RINGHALS KRAFTSTATION AGGREGAT 3 OCH 4 STATIONSPLAN PWR	Ritad 11.9.71. J.L. Kontr. Skala 1:2000 Blad / Forts. bl. / Ritn. nr 3	Granskn. Godkänd Grupp

Ritn. nr