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NUCLEAR POWER PLANT PERFORMANCE IN POWER SYSTEM CONTROL A REVIEW OF INTERNATIONAL PRACTICE

Study Committee 39 Meeting in Toronto,

Canada- September 16-21, 1985

F.L. Carvalho on behalf of Working Group 04

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includes 83 reactors which are in the construction phase with an expectation of commercial operation by 1990 having a gross capacity of 83 GW (79 GWso). Thus the ratio of operational reactors to the overall sample is ~ 78%, and 72% by generation capacity. By contrast, the IAEA reports that at year end 1984 there were 344 reactor units connected to grids in 26 countries having a generating capacity of ~ 219 GWso. Thus, this survey covers ~ 84% of the world's operating power reactors to end 1984, having a generating capacity of 203 GWso or ~ 93% of world operational capacity at the end of 1984.

The unofficial world league table for percentage share of nuclear generation in total electricity supplies as reported by the IAEA to the end of 1984 is as follows:

		% nuclear			% nuclear
(1)	France	58.7	(9)	Hungary	22.2
(2)	Belgium	50.8	(10)	Spain	19.3
(3)	Finland	41.1	(11)	Gt. Britain	17.3
(4)	Sweden	40.6	(12)	USA	13.5
(5)	Switzerland	36.5	(13)	Canada	11.6
(6)	Bulgaria	28.6	(14)	USSR	9.0 (Est)
(7)	F.R. Germany	23.2	(15)	Czechoslavakia	8.5
(8)	Japan	22.9	(16)	Netherlands	5.8

(% nuclear for world ≃ 13%)

By contrast the top 12 users of Electrical Energy in this survey as reflected in the system capacity data of Table 1 for 1985* are as follows:

System Capacity,		GWso	(% nuclear)
(1)	USA	667	(9.5)
(2)	USSR	348	(8.1)
(3)	Japan	174	(12.2)
(4)	F.R. Germany	98	(16.0)
(5)	Canada	91	(10.2)
(6)	France	87	(38.0)
(7)	Gt. Britain	75	(14.7)
(8)	Italy	56	(1.5)
(9)	Brazil	44	(0.7)
(10)	Spain	36	(11.8)
(11)	Sweden	30	(22.2)
(12)	S. Africa	26	(3.5)

* 1985 data based on arithmetic average of capacity data, 1980 and 1990.

Footnote: For additional reading of general interest see page 1.36.

PART 1 : A REVIEW OF INTERNATIONAL PRACTICE

1. INTRODUCTION

In most countries, early nuclear plants were equipped with limited facilities for controlling output automatically in response to system load and frequency. In view of the economic criteria which favours base load operation for thermal plant with lowest production cost, there continues to be an incentive to ensure maximum nuclear generation at all times without disturbance arising from frequency deviations. However, as the installed nuclear capacity becomes a significant proportion of the total, it is necessary, especially at times of light system load, to make provision for load and frequency regulation from the nuclear plant operating in conjunction with or isolated from other generation sources.

This paper reviews the principal reactor systems in commercial use for power generation in 21 countries and identifies the design features of the different types.

The paper is presented in two parts. Part 1 considers the principal design features of nuclear steam supply systems, their control capabilities and provisions for load rejection, islanding, isolated operation, and reviews present practices and possible future trends. It also summarises events during some major system disturbances, and in particular, the impact of such disturbances on the operation of nuclear plant. All the response characteristics presented were obtained from actual events. The development and redesign of control systems for existing plant is also considered as being an on-going requirement for utilities.

Part 2 presents the compilation and analysis of the international survey by CIGRE members covering a substantial sample of the world's NSSS. It covers reactor plant, control systems, load changing capability, and the ability to undergo load rejection followed by quick reloading.

2. COUNTRIES SURVEYED

TABLE 1 lists the data for commissioned nuclear capacity, system capacity and % nuclear component of system capacity for 21 countries and includes non-utility generation for some countries. Generating capacity for these countries based on nuclear fuel sources which is expected to be commissioned and operational by 1990 amounts to ~ 76% of the total world's anticipated nuclear capacity. World projections for end 1990 are 509 reactors in 32 countries having a total capacity of ~ 370 GWso [2] [9,2)]. The data of TABLE 1 is based on the sources referenced [9] and updated by national representatives and correspondents in response to enquiries related to this survey. Fifteen countries were represented in the Full Working Group and data on the remaining six countries was obtained by correspondence with utilities in those countries. A good response was obtained to enquiries initiated via a questionnaire (Appendix A.5).

FIGURE 2 shows the trend for the data of TABLE 1. This shows the growth rate, actual and forecast, in the period 1970-1990. The forecasts for commissioned nuclear capacity taken from TABLE 1, expressed as a percentage of system installed capacity for each country shows the following grouping:

COMMISSIONED NUCLEAR CAPACITY AS % OF INSTALLED SYSTEM CAPACITY *

Range, %	1970	1980	1990
~ 50 and above			France
35-50			Belgium
20-35			Finland, F.R. Germany, Korea, Sweden, Taiwan,
10-20		Belgium, France, Gt.Britain, Japan, Sweden, Switzerland, Taiwan, Finland	Argentina, Canada, Spain, Gt. Britain, Hungary, Japan, Switzerland, USA, USSR,
5-10	Gt. Britain	Canada, F.R. Germany, Korea, USA,	South Africa
1-5	France, Italy, Switzerland,	Argentina, Holland, Spain, USSR,	Brazil, Holland, Italy, Romania,
<1	All other countries	Brazil, Hungary, Italy, Romania, South Africa,	

This indicates clearly the increasing role for nuclear plant in supporting load demand in future years. It may also be assumed that in the earlier years 1970-80, the nuclear contribution to load demand being less than 20%, the role for nuclear plant was essentially one of base load except in unusual circumstances e.g. during very light system loads in countries with substantial water resources and hydro plant [58]. The projections for 1990 however show a significant increase in nuclear component of system capacity for nearly all the countries surveyed over the two previous decades and does suggest a possible need for load-following duty and frequency control by nuclear plant in some countries in the 1990s. These aspects are dealt with later in the report and survey.

* from TABLE 1

3. DESIGN FEATURES OF NUCLEAR STEAM SUPPLY SYSTEMS

The majority of power reactors in various countries belong to one of the following categories:

Water-Cooled Reactors	Light Water	pressurized or PWR
		graphite, channel LWGR
		boiling or BWR
	Heavy Water	indirect cycle HWR
		direct cycle SGHWR ‡
Gas-Cooled Reactors	CO2/Water	low temperature MAGNOX
		high temperature AGR
	Helium/Water	highest temperature HTGR,
Liquid-metal Cooled fast Reactors	Sodium/Water	Breeder reactors FBR,

‡ or BWR-PT

(For abbreviations - See Section 12 : page 1.42)

TABLE 2 shows the main features of the NSSSs. FIGURES 3 and 4 illustrate the steam cycles of the more common systems. For clarity, some of the components have been excluded, e.g. the pressurizer for the PWR, and the Steam/Steam Reheaters for Water Reactor Systems.

3.1 Water Reactors

Water-cooled reactors form the largest group of power reactors being built in the world today being about 93% of total nuclear capacity commissioned and in construction [9, 2)].

The light-water systems involve pressure vessels of massive structure to enclose and contain the primary coolant - light water or H_2O. Water reactors utilizing heavy water, or D_2O, can be one of two types - an indirect cycle, where the primary coolant D_2O exchanges its heat in a secondary exchanger D_2O/H_2O to make steam for power generation, or in a 'direct' cycle - BWR, SGHWR, where the primary coolant is H_2O and is itself the working fluid in the turbine. Indirect cycle Heavy Water Reactors employing Natural Uranium provide an alternative to the use of enriched uranium in PWR, BWR, and GCR's. Two basic designs (CANDU and KWU) are being produced at this time. The Canadian CANDU design [33] uses horizontal thin-walled pressure tubes. The German KWU design [32] uses a pressure vessel as in the PWR. Both are refuelled on load.

An integral part of Pressurized Water reactors is the pressurizer, the dimensions of which are geared to the steady state characteristics adopted for operation (refer Section 5.1 - PWR). A system of electrical heaters and spray water are usually employed for controlling primary circuit pressure [33,45]. Some of the early Russian VVER type PWRs used nitrogen as the pressurizing medium [10].

The Russian water reactors are of two types - the VVER type: light water cooled light water moderated vessel type (PWR-PV) [10], or of the RBMK or pressure-tube type (PWR-PT) [11]. Both reactor types are built in various sizes [1,3]. The RBMK type is also referred to as the channel type or uranium graphite reactor. In this survey it is classified as an LWGR.

All water reactors are fuelled with uranium oxide (UO_2), in the form of solid cylindrical pellets carried within tubes to form fuel elements - TABLE 2. High power density of the water reactors coupled with their relatively small volume and excellent heat transfer characteristics enable fast load ramping to be achieved and this is an inherent feature of such systems.

3.2 Gas-cooled Reactors

Gas-cooled power reactors commissioned in the early 1960's in Great Britain, France [14,15,16], Italy and Japan are of the low temperature Magnox type. FIGURE 4(a)(b) shows the basic features. In France the early designs favoured concrete pressure vessels and later designs changed to steel vessel containment. In Great Britain the reverse was the case. Because of the relatively low gas temperatures, steam generation at two pressures was used except in later designs where once through boilers replaced the drum boilers. Total Magnox generation for these countries amounted to about 8000 MW by the early 1970's.

Gas-cooled power reactors commissioned from the mid-1970's onwards operate at higher temperatures, use oxide fuel in stainless steel cladding, are of the AGR type incorporating concrete pressure vessels and once-through boilers as in FIGURE 4(c). Output projected for AGR stations in the UK is about 8700 MW by the late 1980's.

The High Temperature Reactor is seeing a limited application in the USA and FRG [17,18,19]. The plant layout is similar to the AGR in terms of containment but the primary coolant is Helium instead of CO_2. The use of cogeneration steam cycles is favoured in some designs [18].

3.3 Fast Reactors

Liquid sodium is the primary coolant, and as no moderator is involved, small size and high power densities are possible. Intermediate Sodium/Sodium Heat exchangers are a feature of the design. A few FBRs are in operation in the power generation industry in some countries and experience to date has indicated that such systems have the ability to load-follow [20,21,22 and 27, 69].

3.4 Steam Generators

Steam generators may be classified as one of two types - recirculating or once-through. Both types are used in the indirect cycle, with a preponderance of drum type units for existing designs. In the case of the BWR and SGHWR the steam generation process occurs and is completed within the reactor vessel, there being no secondary coolant or heat exchange circuits.

(i) Drum Boilers

Drum boilers or recirculating boilers have seen application in Magnox plant, PWRs, HWRs - of the KWU and CANDU type, FBR (Dounreay, Scotland). The most common type of steam generator for PWR applications is the vertically orientated U-tube design. In order to contain the pressurized primary reactor coolant, the containment, tube-plates and fluid entry nozzles are of massive structure. Additionally, because of the fast-response characteristic of PWRs and HWRs, the fluid inlet and outlet nozzles have thermal sleeves to reduce the effect of thermal stresses arising from rapid temperature changes. In the case of the FBR, intermediate heat exchangers are used to transfer heat between the reactor unit and the secondary steam generators. Steam generated in such boilers as described above (with the exception of FBR) is saturated and nearly dry. In the case of the FBR some superheating is carried out post-steam generation in separate superheaters [20].

(ii) Once-through Boilers

Once-through, or Benson boilers are used in Magnox and AGR stations employing Concrete Pressure vessels so as to minimise penetrations of the containment (FIGURE 4(b)(c)). This also applies to CPV designs incorporating HTRs. It is also feasible to design once-through steam generators for PWRs and some studies have been reported [23]. The Advanced Thermal Reactor, or FBR located at MONJU in Japan embodies a once-through steam generator, theoretical studies of which have been reported [22]. The study also contrasts dynamic behaviour characteristics of drum and once-through systems for the FBR. The FBR at Creys Malville in France (Super Phenix) also embodies once-through steam generators [27]. In the case of AGRs, because of the higher temperatures of primary coolant compared to Magnox, operating constraints based on material temperatures and coolant state apply [24], but against this the once-through steam cycle can afford a greater flexibility in terms of steady-state regime options and controllability.

(iii) Steam Pressures and Temperatures

Most steam generators for NSSSs operate at pressures below the critical pressure (22 MPa), and at steam temperatures of between 250-560°C. The saturated steam cycle operating at pressures between 3-6.5 MPa and 230-285°C is particularly suited to the Water Reactor family including the Russian VVER and RBMK types. A developed version of the latter which incorporates a measure of steam superheating is termed the RBMKP type and is to be used in the larger reactor sizes of 1200 and 2400 MWe [1].

In the case of AGRs and HTGRs the higher primary coolant temperatures enable better steam conditions to be achieved, and with a substantial degree of superheat. Thus steam pressure of 16 MPa and steam temperatures of 540-560°C are achievable with consequent improvements to steam cycle efficiency. Such steam conditions also apply to the FBRs [20,21,22,27].

(iv) Boilers/reactor - Configuration in NSSS

Most NSSSs embody multiple boilers per reactor with facilities for isolation of boilers. The number of boilers may be as low as 3 for a PWR or up to a dozen or more especially for some NSSSs as in France (early Magnox) and the USSR.

3.5 Steam Reheaters

Reheating of steam after expansion during passage through the high pressure section of the turbine improves the steam cycle efficiency. It also serves the important purpose of removing moisture from the steam which has an adverse effect on turbine performance by worsening turbine blading efficiency and promoting erosion and crevice corrosion of blading [28,29,64,70,71].

Broadly, the provision and manner of steam reheating in NSSSs is governed by the reactor system and the primary coolant conditions. It can take one of two forms when such provision is made.

(i) Reheating by the primary Coolant

For NSSS employing higher temperature coolants e.g. AGR, HTR, and FBR, it is usual to reheat the steam leaving the HP Turbine Section by returning it to the Reactor System. This results in a reheater arrangement which is integrated into the reactor heat removal system FIGURE 4(c). In its impact on control dynamics, such a system introduces longer settling times particularly for gaseous coolants.

(ii) Reheating by live steam

For NSSS employing saturated steam cycles and hence lower primary coolant temperatures and secondary steam pressures and temperatures, it is common practice to provide HP exhaust steam reheating using HP admission or HP bled steam as the reheating agent. Later Magnox stations (FIGURE 4(b)) PWRs, BWRs, and other water reactor systems (as in FIGURE 3) make use of such arrangements.

Stored Energy in Reheaters prolongs plant reaction times and potentially worsens the overspeed characteristics of steam turbines during load rejections. Special arrangements are made to protect against this risk, for example by the provision of Interceptor Valves at the IP/LP Turbine inlets and by other means [25,26].

3.6 Steam Turbines

The design of Steam Turbines for use in Nuclear power stations depends on the steam conditions available from the Steam Generator, and this in turn depends on the reactor system and primary coolant conditions. Basically, two types are in use:

(i) Superheated Steam Turbine

Such turbines follow design practice as for fossil-fired stations and usually embody an external reheater (integrated into the reactor system as stated in 3.5(i)). Applications are AGR, HTR, FBR and RBMKP water reactors, and for somewhat lower temperatures, Magnox.

(ii) Saturated Steam Turbine

Turbines based on the saturated-steam cycle have greater proportions than the superheated steam temperature machines described above. The valve gear and pipework are also of greater dimensions. Because of the presence of entrained water throughout the machine and its associated pipework, special provisions are made to reduce or eliminate condensed vapour. Any water or water vapour present in the steam will, during negative steam pressure excursions, flash to steam tending to overspeed the machine. Such pressure excursions would result from primary governor action, but in the worst case from a load rejection. The defence provisions which are made to accommodate and minimize these effects are described fully in ref. [25] and referred to elsewhere [26]. Applications are BWRs, PWRs, HWRs, VVER and RBMK reactor systems.

(iii) Special 'Power + Process Steam' Turbines

NSSSs embodying turbines for power generation + process steam for district heating feature strongly in future plans of the USSR. A combined heat and power plant is in construction near Odessa, and is based on a VVER - 1000 reactor (PWR of 1000 MWe capacity). Future 5 year plans embody further proposals along such lines [1,10].

Limited information is available on plant configuration for Russian stations. However this survey contains material data, presented in the Database on the control arrangements for NSSS embodying VVER reactors used exclusively for power generation [31].

(iv) Turbines/reactor - Configuration in NSSS

Early nuclear plants embodied more than one turbine per reactor. This applied to Magnox plant in particular. Current designs often favour only one turbine per reactor, and the survey provides information on individual plants. Nearly all of the Russian plant utilizes two turbines (more on early designs) per reactor. This appears to be a special feature of their standardized approach. The AGRs follow the normal pattern for PWRs and other water Reactors, namely one machine per reactor. The availability of more than one turbine per reactor does afford a measure of operational flexibility during turbine-generator outages.

3.7 Turbine By Pass Systems

The provision of a Steam Bypass system is necessary to a NSSS to accommodate an interruption to the consumer demand, or operation of the turbine or generator protection system which may arrest, partially or totally, the passage of steam to the turbine. The provision of bypasses can take one of two forms, or both together as follows:

(i) Atmospheric Safety Valves

These valves are the Safety Valves which are located at positions in the Secondary Steam Circuit. They limit pressure build-up following steam valve closures on the turbine. They are usually located on the Boiler HP Outlet and/or the HP Turbine main steam header, and at the Reheater Outlet and/or the IP/LP Turbine inlet crossovers. The valves are sequenced to open at preset values of steam pressure.

All NSSSs are provided with these or similar arrangements. Losses of steam require to be made up by the reserve feedwater systems.

(ii) Steam Venting to the Condenser

Many NSSSs are fitted with Bypass Systems - which, via pressure reducing and cooling arrangements, permit the discharge of unwanted steam to dump condensers within the main turbine condenser [34]. The Bypass valves and pipework are sized to accommodate certain levels of flow and pressure in a range from about 10% to 110% of MCR values. The discharge routes are controlled by a pressure measurement local to the Bypass Valves. Depending on the disturbance level and its persistance, and the capacity of the Bypasses to the Condenser, operation of the Atmospheric Safety Valves may also occur during load reduction. Generous provision for Bypassing steam to the condenser enables steam (and hence feedwater) to be conserved during disturbances and power excursions which affect the turbine/generator(s).

An example of a steam bypass to condenser arrangement and safety valve array for an AGR at Dungeness 'B' power station is shown in the inset to FIGURE 26.

(iii) Implications for Reactor Operation

Steam Bypass provisions in the form of Atmospheric Venting or Dumping to Condenser, enable smooth transitions in reactor loading to be achieved automatically or manually, following a major change in turbine steam flow.

For short-lived disturbances, e.g. 5-10 minute disconnection of a NSSS from the grid, provision of the Atmospheric Vents would be adequate without incurring great losses of steam and water from the system, and reactor operation could be maintained at the original load without disturbance to control rods or absorbers.

For longer disturbances, loss of boiler steam and water and the extent of the Reserve Water capacity will determine the maximum run-through time pending reconnection. Water losses can be reduced by reducing reactor load for an extended run-through pending a reconnection (as actually occurred during the loss of grid connection for 5h 35m during islanding of a Magnox NSSS - Section 5.1, FIGURE 22). However extended venting of steam to atmosphere may not be allowed by local authorities in some countries.

Load reduction for extended periods coupled with Safety Valves (above) venting to Atmosphere may necessitate a reactor shutdown due either to loss of Boiler Water Reserves, or 'poisoning-out' or both. Such situations can be eased by the provision of turbine bypass systems with a capacity adjusted to the reactor set back capability.

(iv) Implications for Turbine Run-Through

Besides affording conservation of steam and water in the secondary circuit and enabling continued reactor operation without imposing sudden change, the provision of additional dumping of steam to the condenser reduces the overspeed of the turbine during large load rejections. This is of particular relevance to large capacity saturated steam turbines as discussed in Section 3.6 (ii) [25,26].

3.8 Refuelling Operations

The operation of refuelling a reactor, either partially or wholly, may require it to be shut down or incur a reduced loading for the duration of the operation. Most PWRs and BWRs require to be shut down for a period of several weeks every 1-3 years depending on the fuel cycle adopted together with the absorber inventory in use. The HWRs (KWU and CANDU) are refuelled on-load, and this is carried out almost daily, a few bundles at a time.

Of the Russian reactors, the VVER type (PWR) can only be refuelled off-load, partial refuellings once a year with a 30 day outage. The RBMK-1000 type uranium-graphite channel-type reactor cooled with boiling water is refuelled on-load with several hundred refuelling operations per year being necessary [1]. Spent fuel from a VVER reactor can be used in an RBMK type to achieve further burn-up.

Magnox and AGR reactors can be refuelled on-load.

4. CONTROL CAPABILITIES

4.1 Response Duties imposed by the System

In considering plant response characteristics, it is appropriate to review the requirements imposed by needs of typical power systems. In several articles published on the subject, the response duties for generation have been related to those imposed by system electrical load demands and by dynamics of electrical grids under normal and emergency conditions [39,40,41,42,43,44,46]. These response duties are categorized as follows:

(i) Daily load following (cycling)

In order to meet system demands, generation must have the ability to change load daily over a wide range, at rates of 1% to 5%/min, which are the normal maximum rates for generation management or automatic dispatch system controls. To provide wide operating flexibility, the load range should be typically from 40% or less to Full Load. This is well within the capability of many NSSSs available today.

(ii) Normal system frequency regulation

Under normal conditions of power system operation, the fast random variations in grid frequency which occur on a minute to minute basis as a result of random changes in load demand are limited by primary speed control action distributed among many generating units connected to the system. This regulating duty on any given unit is minimized by having generating units share the action which they would do if their speed regulation and response characteristics are similar.

On large interconnected systems such as in the US Canada and Western Europe, the fast random deviations in frequency are so small that during normal grid operation the resulting primary speed control action is imperceptible. For instance, the band of fast frequency deviations in the major US and Canadian interconnected systems seldom exceeds ± 0.03 Hz. With typical governor droop settings of 5%, the corresponding primary speed control duty would result in power variations of merely ± 1%.

FIGURE 5 shows the cumulative distributions of frequency and time error based on values recorded every 10 seconds at Ontario Hydro's System Control Centre for the Eastern Interconnection of North America (c.367 GW peak demand) for the period 1981-3 [61], and for the UK system (c.40 GW peak demand) for 1982-3 using 30 minute averaging. The target limits for the North American Systems are ± 0.5Hz or 0.83%. In the UK the figures are ± 0.5Hz, ± 1% (statutory limit) and ± 0.2Hz, ± 0.4% (operational limit) for the CEGB's system. Also shown in FIGURE 5 is the data for the West European Interconnected Grid (c.150 GW peak demand) for 1977/78, derived from source information in reference [43].

On small grids however, as in many developing countries, large frequency variations are more common, ± 1Hz for example.

(iii) Grid frequency recovery following a large disturbance

Another consideration is the need for a unit to respond quickly to unusually large frequency deviations under speed control action, i.e. in 10-20 seconds. However, this occurs very infrequently in large interconnected systems. For instance, in the US, the loss of a 2000 MW power plant causes a drop in frequency of the order of 0.05 Hz in the Eastern Interconnection of North America. In such systems a major imbalance between load and generation could only occur during a severe disturbance resulting in system separation or islanding. For nuclear plants in the F.R. of Germany the capability for an increase of 5% of rated load in 5 seconds is required when Germany is disconnected from its neighbours in the West European interconnected system, instead of 2.5% in 5 seconds without separation - FIGURE 6. In the case of grid disturbances that cause a separation of the plant from the grid, the unit has to supply its own houseload and must not trip* [43].

On smaller systems where individual units or plants can represent a significant fraction of the total generation the incidence of events that would cause large frequency deviations could be much more frequent. Spinning reserve in the form of steam, hydro or gas turbine generators would in such cases be called upon to respond under governor action. NORDEL recommends achievement of a 10% step in 30 seconds for BWRs [44]. If power systems are comprised of a large proportion of NPPs, spinning reserve would have to be provided in nuclear units and similar fast response capability would be required.

(iv) Nuclear plant flexibility to control transmission system overloads

Nuclear plants may be required to reduce power within minutes to control overloading of transmission lines or transformers. These transmission system elements have time constants of the order of 15 minutes, and, accordingly, reduction of power output at rates approaching 5% per minute should be adequate, but this is expected to occur very infrequently. However, as demonstrated in Section 5.3, the fast response capability of nuclear plant can easily accommodate such requirements.

(v) Operational Performance Specifications for Units

There exists no single standard specification of requirements for nuclear plant. The practices vary according to country of operation and are therefore a matter of individual application.

The broad guidelines for the Scandinavian countries, Norway and Denmark (no nuclear plant), Sweden and Finland, are set down in the form of recommendations of the Thermal Power Committee of NORDEL [44]. The installed capacity of the system is ~ 60 GW with a maximum load of ~ 40 GW. The recommendations cover load ranges and loading rates for specific classes of plant e.g. fossil-fuelled, PWR, BWR, and a general category 'nuclear', and there is a brief reference to 'islanding' - see 5.4 (ii) (f).

* This is also a requirement of NORDEL - see 4.5 (ii) (f).

Guidelines for the Federal Republic of Germany, grid capacity ~ 95 GW, are contained in two DVG publications [43,46]. FIGURE 6 illustrates the requirements for Primary Reserve in the FRG. Data based on model studies of the West European Interconnected Grid System UCPTE* using simulation techniques are used to develop the recommendations put forward. No distinction is made for class of nuclear plant such as 'reactor type' as for the NORDEL recommendations.

Examples of contrasts between the operating criteria adopted by NORDEL, North America, and UCPTE in respect of security and quality of power supply are given in a recent paper presented to CIGRE SC39 by Hagenmeyer on behalf of Task Force 01 [47].

Additionally, the control capabilities and interaction of grid and nuclear plant is given in an IAEA publication [48]. It includes auxiliary systems in nuclear plants.

4.2 Nuclear Plant Control Philosophies - TFR, RFT, CC

In view of the low unit costs of nuclear-based generation, most countries operate nuclear plant in a base-load regime. Accordingly, in these cases the system requirement for power cycling and frequency regulation is not a prime consideration, especially for the early NPPs which do not incorporate automatic control systems with some exceptions. For example, in Germany, utilities specified auto control systems in anticipation of load-following duty at some later date [46,72,73].

The philosophies for the control of nuclear plant fall into one or both of two categories as follows:

(i) Primary control - speed governor in control range

(ii) Secondary control - power adjustment with load following

The first of these, because of the relatively fast speed-change characteristics of the rotating components in the turbine-alternator system, is mandatory for safety reasons in the closing direction of the turbine throttle valves, whilst the second is an optional scheme, but one which may be required in the future.

The strategy for implementation of the philosophy (ii) illustrated in FIGURE 7 can take one of three basic forms as below:

(a)	Reactor Following - Turbine (RFT) **	power controller operates on turbine control valves, with follow-up controls on reactor
(b)	Turbine Following - Reactor (TRF) #	power controller operates on reactor with follow-up control of steam pressure by adjustments to the turbine control valves
(c)	Coordinated Control- (CC)	a combination of (a) and (b)

* UCPTE : Union for the Coordination of Power Generation and Transmission of Electricity (in the West European Interconnected System)

** Sometimes also refered to as a 'coupled' mode.

\# Sometimes also referred to as a 'de-coupled' mode.

If the plant is required to provide power dynamically for support of grid frequency, the turbine throttle valves are adjusted by the turbine speed controller - (a) above. The steam pressure fluctuates and the turbine draws upon the stored energy in the NSSS that is, from the steam generator (indirect cycle) or from the reactor (direct cycle, BWRs). In large systems with total capacity exceeding ~ 100 times unit capacity this type of duty is small if all units share in primary control.

If the plant is not required to contribute its stored energy to the grid under primary control action, or if such a contribution is to be limited, action in response to frequency change can be prescribed by the use of one or more of the following means:

- Operation with the steam valves against a load limit negating movement in the opening direction. Excursions in the closing direction due to overfrequency may be inhibited to the extent of the margin between the load limit setting and the normal valve position demanded by the load reference.

- Use of a deadband in the speed controller *

- Use of a deadband in the frequency/load characteristic which corrects the set value of the load controller within a limited load band. That determines the valve position and the output directly in addition to the speed controller as used in German plants [25,49].

- Use of turbine throttle valves to control steam pressure as in (b) above with an adjustable bias from speed or frequency deviations.

Hybrid systems which adjust simultaneously the reactor, boiler, and turbine systems in response to loading commands - sometimes referred to as 'integrated controls' or 'co-ordinated control' -abbreviated CC - still retain the identity associated with (a) or (b) depending upon whether provision is made for adjustment of boiler steam pressure by turbine valve movements, system (b) or by other means system (a) - see FIGURE 7.

When making provision for secondary control, it is usual to include an option for disengagement to enable constant power operation of the NSSS. A common practice is to operate the plant at full load with minimum influence from system disturbances through use of a load-limiting feature of the unit controls.

Where it is desired to operate nuclear plant to a base-load and insulate it from system fluctuations, use of a Turbine-Following Reactor control philosophy is often used. However, in order to accommodate load rejection or tripping to houseload, it may be desirable to arrange control system switching from the TFR mode to RFT to reduce reactor power quickly and secure the advantage of continued reactor operation. (See also Section 6.1 (iv)).

Control system designs for Russian Water Reactors enable either strategy to be adopted depending on the operating regime i.e. TFR for base load and RFT for load flexing duty [31]. FIGURE 8(b) shows the outline of the unit power control system as arranged for load following duty (= RFT). For constant load operation, the steam pressure signal is routed to the Turbine Load regulator and the frequency signal (f)

* See Section 4.6 for amplification of this provision.

is disconnected from the control system. Here, the loading level is set to a fixed value by use of the Set Point (Command) 22, (= TFR). FIGURE 8(a) (i) or (ii) shows the choice of steady state schedules for TFR mode, and (iii) for load-following, RFT mode for the VVERs.

4.3 Loading Rates and Load Changes

Loading rates generally depend on NSSS type design, control concept, operational mode, etc. The loading rates vary from 0.5% per minute for early nuclear plant to 60% per minute for more recent plant and is dependent on the magnitude of the change.

(i) Gas Cooled Reactors

Loading rates for gas-cooled reactors range from 1%-10% MCR m^{-1}, the very early plants having the lower rate. Many of the early designs could not withstand sustained load reductions greater than about 50% because of poison build-up if allowed to continue beyond a few hours. This could also affect the islanding performance of these reactors on an extended timescale.

(ii) Water Cooled Reactors

Water reactors can in general accommodate much faster rates of change of load because of the higher heat transfer rates, higher power density, and lower thermal inertia of the primary and secondary coolant loops. To illustrate this, the transient response characteristics of the principal parameters of a 1300 MW PWR are shown in FIGURE 9 for 0-15s, and 0-120s. The new value of generation is seen to be established in 5-10 seconds. For PWR's, FIGURE 10 shows typical rates of load change and the relationship with magnitude of load change where the changes are effected in one manoeuvre. Detailed results obtained from tests carried out on five NPPs in the FRG are given in reference [50].

(iii) Fast Breeder Reactors

The large thermal capacity of the sodium loops in FBRs does not restrict their load-changing capabilities. The three FBRs at Dounreay (Scotland) [20] Creys Malville (France) [27] and Monju (Japan) [22] are each controlled in TFR mode although Dounreay was designed for RFT*. Ramp rates on Sodium Flow are 17% and 7% per minute for Dounreay and Monju respectively whilst Creys Malville is designed to accept ± 10% step demands on Sodium Flow. The FBR at Kalkar in the FRG is controlled in the CC mode.

(iv) Pellet-Clad Interactions (PCI)

Rate of change of load can be influenced by the need to avoid pellet-clad interaction (PCI) problems [50, 51, 52, 12]. This is also described in a previous paper by WG 04 [42].

PCI is due to clad straining caused by: differential thermal expansion, fuel swelling, wheatsheaf ridge height growth and stress concentration over fuel pellet radial cracks, at a time when the clad ductility is low due to embrittlement.

* Creys Malville, or Super Phenix, is designed for CC, but with pressure control on the throttle valves and base load operation it would equate to TFR.

PCI was a common cause of fuel failure in the 70's especially in BWRs. The presence of gaseous fission products and excessive strain in the cladding might fulfil the required conditions for the initiation of a stress corrosion crack (SCC). The strain is a result of the mechanical interaction between the pellet and the clad due to differential thermal growth which would occur during a power increase (i.e. as a result of a temperature increase). The strain of the cladding and the resulting stresses is determined by the clad properties and by the pellet expansion after the point of contact i.e. after the as-built gap has closed. This is also a function of the preceeding power history of a fuel rod.

In order to avoid fuel failures, it is important to limit transient overshoots (locally) of power density in the core [12]. The resulting restrictions on load-follow operation thus depend on the capability of the plant to maintain constant power distribution under non-equilibrium xenon conditions. This is achieved in PWRs [45,72] by effective power control and limitation systems. In BWRs it is normally achieved by restrictive operating rules. Considerable efforts are being made to eliminate these restrictions by fuel design changes or improved control strategies [51]. Tests have been carried out at Brown's Ferry to evaluate design and operational strategies to improve understanding of the problem and help eliminate constraints [52].

4.4 Provisions for 'islanding' of nuclear plant

The term "islanding" is used to describe the situation where one or more plants are suddenly subject to a large change in system frequency, usually caused by a separation of parts of the grid from a main grid.* This usually results in frequency excursions and voltage fluctuations. It should be noted that in an "islanding" situation, the plant is called upon to change its output under governor control of its main steam admission valves. To supplement this (though it may not be considered to be necessary by some manufacturers and utilities) additional automatic control features may be provided to assist with the control of the NSSS.

The provision for 'islanding' of nuclear plant varies from country to country, and also within utilities of a country and is influenced by the following practical considerations:

(i) The configuration of the grid connections local to a power station and its switching flexibility and reliability with respect to main grid export and local load.

(ii) The probability of disconnection of the main export component of generation based on the vulnerability of the connection to the main grid in the local environment of the station and the grid lines in the vicinity [66].

(iii) Local legislation or nationally agreed regulations permitting, or otherwise, the operation of nuclear plant when separated from the main network.

Thus, given that the operating rules allow it, the specific provisions made in the design stage of the plant will be governed by the first two considerations and may comprise, or be influenced by, one or more of the following:

(a) Tripping to Houseload (TTH)

* See also Section 6.

(b) Turbine by-pass systems

(c) Above average reserve feedwater capacity

(d) Wide-range auto-control systems

(e) Houseload supply concept

(f) Criteria for the separation of the NPP from the Grid

A companion paper by the Working Group dealing with TTH lists the provisions in various countries outlining the philosophies adopted [35].

It should be noted, however, that following the construction and operation of a station, the development of local industry may require modifications to the grid connection near a station, and it may, as a consequence, be desirable to review the design philosophy in respect of islanded operation of a station*. Also, the benefit of operating experience will enable continuing assessments to be made of the reliability of the main grid connection, including the reliability of the transmission network in the vicinity which carries the bulk station export [66].

Also of great importance in this context is the role played by operations staff faced with unusual operating conditions such as may be experienced during loss of grid-connection. Experiences from real incidents need to be documented comprehensively to enable source information to be accumulated for design and operator training purposes.

Such conditions can be triggered by plant failure, adverse weather conditions - conductor-galloping in high winds and near-freezing conditions, storms, extreme heat leading to excessive conductor droop and subsequent flash-overs to trees and so on.

Some practical experiences are given in Sections 6 and 7.

4.5 Tolerance to Power Supply Variations

An understanding of the effect of voltage and frequency variations, both transient and sustained, on the main plant items e.g. the generators and transformers and on the major auxiliaries i.e. the pump motors and control systems is an important aspect affecting the reliability of an electricity supply system. There is also the associated interaction between load, voltage and frequency which influences the capability and behaviour of main plant items and auxiliaries, transiently and for persistent disturbed grid conditions which needs to be considered and provided for. Some guidelines exist by way of recommendations (referred to earlier) - by NORDEL [44] and DVG [43,46] and in addition, British Standard 5000.

In practice, continued plant operation during and after grid electrical disturbances depends on many factors besides robustness of auxiliaries to supply variations. Design measures are usually taken to secure guaranteed supplies backed up by batteries and so on. Also, a key factor in such situations is the important contribution made by operating staff.

* refer also to Section 8.1

(i) Tolerance to voltage variations

A summary of the NORDEL recommendations are that:

(a) unit rating should be achievable at a generator terminal voltage in the range 90-105% of the rated voltage within a frequency range of 49-51Hz, continuously.

(b) at least 90% of unit rating should be achievable for generator terminal voltage in the range 85-90% of the rated voltage within a frequency range of 49.7-50.3Hz, for a period of one hour.

(c) near full output should be achievable at a generator terminal voltage in the range 105-110% within a frequency range of 49.7-50.3Hz, for a period of one hour.

(d) the plant should be capable of withstanding a step reduction to 25% of the rated voltage of duration 0.25s, followed by a linearly increasing voltage to 95% within 0.5s, followed by a sustained voltage level of 95% of rated value, without significant power reduction or disconnection from the grid.

(e) for voltage reductions greater than the above, the unit should be disconnected from the network. The unit and auxiliary power system should be designed for such voltage variations as to enable a safe change-over to operation on its own auxiliary power following disconnection.

(f) the generator voltage control system should be designed to enable the generator to provide a reactive power output of the same magnitude as the rated active power output for 10s during network disturbances at a generator terminal voltage of 70% of rated value.

The BS 5000 guidelines broadly are:

(g) unit rating should be achievable within a voltage range 95-105%, and within a frequency range 49-51Hz, continuously - as for (a).

(h) reduced unit output should be possible within a voltage range 95-103%, for a corresponding frequency range 47.5-49Hz, or 95-105% volts in the frequency range 51-51.5Hz.

(ii) Tolerance to frequency variations

Some of the requirements of plant performance in the presence of frequency variations are covered above under (i) in the case of the NORDEL recommendations. The other recommendations stated in [44] are:

(a) Within the generator terminal voltage range of 95-105% of rated voltage, and within the frequency range 51-53Hz, it should be possible to operate at greatly reduced output on a separate network for 3 minutes.

(b) A maximum operating time of 10h/year and a duration of 30 minutes maximum per event can be assumed within the frequency range 50.3-51Hz.

(c) Within the generator terminal voltage range 90-105% of rated voltage, it shall be possible to operate the unit under certain disturbance conditions for 30 minutes at any frequency down to 47.5Hz. The power may then be reduced by 0% at 49Hz and a maximum of 15% at 47.5Hz.

(d) Within the generator terminal voltage range 95-105% of rated voltage, operation should be possible at a frequency between 51-52Hz for 5s.

(e) The unit may be tripped from the network at frequencies below 47.5Hz.

(f) The unit will not trip as a result of the transient frequency fluctuations occurring in the event of short circuits in the grid [implied requirement for TTH].

The DVG documents [43,46] relating to practice in the FRG states that:

(g) Turbosets are designed for continuous operation in the frequency range 48.5-51.5Hz.

(h) Operation of turbo-sets is also permitted for short durations at lower frequencies down to 47.5Hz.

(j) Generating sets are tripped automatically when the frequency falls below 47.5Hz.

(k) For transformers in continuous operation, the frequency range is given as 50 ± 1.25Hz. A summary of the DVG requirements is also included in reference [47].

Russian VVER reactors are capable of operation for short periods at a frequency of 46Hz [10].

Following the generation deficiency events which affected the major North American networks in 1977 - frequency fell to 58.84Hz and persisted for 6 hours in the January incident, EPRI sponsored a study to determine the range of underfrequency that could be tolerated by the interconnected power systems in the US and Canada during a severe loss of generating capacity [39,40].

The systems are:	Estimated Peak Load, GW	1984 Bias Setting MW/0.1Hz
Eastern Interconnection	367	5609
Western Interconnection	99	1898
Texas Interconnection	37	729
TOTAL	503	
Quebec Interconnection	23	250

The first report [39] explored the static effects of under-frequency and further work on dynamic effects was planned. The report concludes that within the frequency band 60 ± 0.5Hz, plant is not at risk. At frequencies lower than 59.5Hz automatic load shedding would be activated. Allowing for the self-stabilizing effect of load reduction with falling frequency (estimated to be -0.22% load per -0.1Hz) operating limits of 59.7, 59.8 and 59.9Hz are recommended for the Eastern, Western and Texas interconnections respectively.

It should be noted that as the Hydro-Quebec system is asynchronous with the rest of North America it has to be able to cope with larger frequency deviations, and it accommodates this by load shedding.

In the Ontario-Hydro system a total of 50% of nominal load may be tripped by low frequency or frequency trend relays (FTRs) in four blocks as follows:

A Guard relay operates at 59.5Hz Below this value FTRs operate at specified rates of frequency decline to reject given percentages of load:

T1	0.4Hz/sec,	10%	T3	2Hz/sec,	10%
T2	1Hz/sec,	15%	T4	4Hz/sec,	15%

T4 is also tripped direct if the frequency falls to 58.8Hz.

In the CEGB system, deloading of 2% of current load per 0.1Hz over 50Hz at selected power stations is carried out with an 'information only' alarm at 50.4Hz. In the lower frequency regime, selected gas turbines start-up automatically at 49.7Hz and load to maximum capacity. This is repeated at 49.6Hz. At 49.5Hz, remaining GTs are started manually with a Low Frequency information alarm at 48.8Hz. Thereafter load is shed in 4 Blocks via low frequency relays which are set as follows:

B1	48.5Hz,	10%	B2	48.3Hz,	10%
B3	48.0 Hz,	10%	B4	47.6Hz,	10%

Below 47.6 Hz, independent manual load-shedding is applied to avoid frequency falling below 47Hz.

In Germany, DVG published a 5-step load-shedding plan in 1980 as follows:

Step 1	49.5 Hz Alarm	Step 2	49.0 Hz 10-15%
Step 3	48.7 Hz, 10-15%	Step 4	48.4 Hz, 15-20%
Step 5	47.5 Hz, Disconnection from Grid.		

Note that in contrast to the Ontario Hydro System strategy the use of FTRs based on rate of change of frequency is not used in the United Kingdom, FRG or a number of other countries.

4.6 Control of Turbine Speed

The use of deadband in the primary speed control system for turbines was referred to in Section 4.2. In view of the need to have reliable control of system frequency and at the same time schedule the control duty to be shared between different generation sources, it is useful to review the requirements in some depth.

(i) Frequency deadband

The recommendations of European organizations in respect of frequency deadbands is as follows:

DVG in references [43,46], max deadband > ±50mHz, NORDEL in reference [44], max deadband not specified. The insensitivity requirement for the frequency controller is specified as being ≤ ±5mHz, (NORDEL and DVG).

(ii) Low frequency operation of turbines.

A number of reports on the implications of low frequency operation of turbo-sets suggest that some manufacturers recommend time limits for sustained operation of their machines at speeds which are removed from the design speed [39,40,48,53,62] to eliminate the risk of failure.

The risk level is based on a fatigue life usage by virtue of high fatigue stresses induced by blade resonance at specified frequencies in the range 0.5Hz - 4.0Hz below design machine frequency. Whilst under frequency protection is favoured and recommended by some manufactures - refer FIGURES 11,12, it is still generally not specified by some utilities or other manufacturers. For example, the CEGB and British turbine manufacturers do not regard it as a requirement [54]. From the point of view of grid system operation, the tripping of generators during low frequency excursions, whether transient or sustained, will accelerate frequency and voltage decline in a generation deficient grid. The strategy of tripping consumer load is widely practised to alleviate frequency falling below acceptable levels so enabling continued grid operation and affording protection to plant and other consumers.

In so far as the generator is concerned underspeed running leads to an increase in flux levels with consequent increase in end-winding heating. There may also be vibration problems, but these are usually a secondary consideration [54].

Failures of blades in the LP sections of steam turbines has cost the utilities c.$270m per year as reported by EPRI in 1980 and 1982 following workshops on this problem [28,30]. In another report in August 1981 [29] prepared by Westinghouse and others on behalf of EPRI, the design aspects and stresses to which LP turbine blades are subject in service, and their failure statistics, are presented and discussed. The report states that the first objective of blade design is to avoid natural resonance by tuning the blades through the first three or four modes of vibration to avoid harmonics of the operating speed. Further, the operating window within the 60Hz (3600 rpm) range between harmonics must be sufficiently removed from the harmonic so as to avoid resonance of individual blades that deviate from the mean as a result of manufacturing variations. Thus resonance conditions are ruled out as a factor influencing blade reliability.

Data presented in the EPRI Survey indicates that most of the blade failures occur in the machine at the transition from dry to wet steam where corrosive impurities may deposit on the blades. In the UK a survey of blade failures has been carried out by the CEGB covering the period 1970-1983 [64]. In this survey it was found that 72% of the failures were in the LP section, with 14% in the HP.

The great majority of the failures are attributed to fatigue alone (77%) or to a combination of fatigue with other mechanisms (16%). One of the findings is that there has been a diminishing trend in the number of failures over recent years (1978-84). This could be attributed to a reducing number of machines, (590 in 1970, 340 in 1977, 155 in 1983) over the period examined due to the progressive decommissioning of older plant. However, the failure statistics of turbine blading also shows that the number of failures 'per set' averaged over all sets, increased by 70% over the early period 1971-1978. During this period and before, the control of grid frequency in the CEGB System was considered to be good. In the later period 1979-1984, the number of failures per set, averaged over all sets, decreased by 10%. During this period the variations in grid frequency have increased. Also, with the demise and decommissioning of the smaller and older units there has been a preponderance of larger units ⩾ 500MW in service in the last 5 years. The failure statistics also show that for this class of plant the mean failure rate per set has steadily declined in the period 1970 to 1983 by 50%, whilst the efficiency of control of grid frequency has also worsened in the latter part of this period.

On the basis of this evidence and the lack of supporting information to the contrary, the risk level associated with the sustained and transient underspeed running of turbines within a few percent of design speed remains to be demonstrated. This aspect may be quite important in small systems which are prone to large frequency deviations. Considerable research effort is being directed to the topic of turbine blade vibration, its measurement, prediction and avoidance.

A recent paper [55] presented at a Symposium on turbine blade vibration phenomena concludes that due to the great number of influencing parameters it is impossible to date to calculate the dynamic stresses and to predict the vibratory behaviour of coupled steam-turbine stages.

5. EXAMPLES OF NUCLEAR PLANT PERFORMANCE CHARACTERISTICS

The following section considers the more important steady state and dynamic characteristics of the more common types of NSSS in the context of load and frequency control, and of the main auxiliary systems during grid disturbance events.

5.1 Steady State Characteristics

(i) PWR

The NSSS consists of a light water reactor in which the operating pressure is chosen such that no bulk boiling occurs within the primary coolant system. There are variations in the specific systems available but typical steady state characteristics are shown in the part load diagram of FIGURE 13(a) [57]. In the upper load range the reactor is operated with a constant value of average primary coolant temperature. This schedule results in a small pressurizer design and minimisation of reactivity changes via temperature effects. For systems incorporating steam turbine bypasses, the bypass set-point value may be characterized according to an inverse sliding pressure schedule to match the main steam pressure as shown in FIGURE 13(a). Such a schedule incurs steam dumping to the condenser during rapid load reductions. The reactor output is controlled with control rods for fast load changes and by changes to boron concentration for small load changes. Xenon build-up and burn-out can also be used for daily load-following duty [72]. Coolant temperature changes help by virtue of strong negative reactivity feedback at the end of the fuel burn-up cycle especially at part load [57]. Steady state characteristics and control system for a KWU-design PWR (Grafenrheinfeld) is shown in FIGURES 45 and 46.

The Russian PWR, denoted VVER, is built in various sizes up to 1000MWe. The early reactors were designed as base load units and were operated to the steady state schedules shown in FIGURE 8(a). The pressurizer size for the type (ii) schedule is greater than for the type (i) schedule owing to the changes in primary coolant density that have to be accommodated. Later VVERs however operate to the schedule shown in (iii) so as to obtain better dynamic response characteristics during primary and secondary control in the load range 75-100% [31]. The plant configuration which includes full capacity steam bypasses to the condenser is also shown in FIGURE 8(b). Experience gained in the design and operation of these units is given in reference [10].

(ii) BWR

The primary characteristics distinguishing the BWR from other types of NSSS is that the coolant circulating through the reactor is allowed to boil in the core creating the steam which drives the turbine. The whole process occurs within a single pressure vessel - FIGURE 3 [51].

The steam generation in the BWR is controlled primarily by two means: control rods and recirculating core flow rate. Steam generation increases as control rods are withdrawn and/or as reactor circulating core flow is increased. Power changes resulting from changes in recirculating core flow are distributed evenly over the core and this method is the first choice for accomplishing load changes. It is effective over a limited range (about 40% of the base power for the particular control rod disposition). For larger changes in power, control rod motion is used to supplement the action obtained with recirculating flow control. FIGURE 14(a) shows typical BWR power-flow characteristics.

The Russian, Light-Water graphite moderated reactors denoted RBMK are also being built for wide application as flexible units of large capacity in the USSR [1,11]. In the survey these reactors are classified as LWGR. The operating characteristics are similar to the VVER type [1] but no specific data is available to describe the steady state schedule.

(iii) GCR

The part load diagram of a typical AGR is shown in FIGURE 15(a). Because of the use of once-through boilers within the concrete pressure vessel, the gas flow and temperatures are reduced with load to conserve auxiliary power and maintain metal temperatures within prescribed limits [24]. The steam superheat at the evaporator outlet is maintained at between 50-100°C to avoid stress corrosion problems in the superheaters. This influences loading rates nominally 10% min^{-1}, and reactor refuelling strategy. Sliding pressure operation is possible but is not envisaged for UK plant within this century.

(iv) HWR

The natural uranium heavy water - cooled and moderated reactor exhibits a small power coefficient of reactivity which allows large stable power changes to occur with only a small accompanying change in reactivity. The controlled reactivity changes are made by simple liquid and/or solid absorber rod systems. The part load diagram - FIGURE 16(a) shows the constant pressure and increasing main coolant temperature feature of this type of reactor. Power reductions give rise to slow Xenon transients developing over ~ 8 hours which, if left uncorrected by control action, can lead to 'poisoning-out' or shutting down of the reactor for large power reductions. Typically, reactors can reduce output to ~ 60% power and remain there indefinitely bypassing steam to the condenser, or can stay below 60% for a limited period (e.g. 30 minutes at zero power or about 60 minutes at 40% power) after which power needs to be increased if 'poison-out' is to be avoided [32,58].

(v) FBR

In the liquid metal FBR, the Sodium Flow is directly proportional to the load in a range of about 30-100%, with the final steam temperature controlled to a constant value of between 480-540°C, superheated with respect to an operating pressure of between 13-18 MPa at turbine admission. Steady state performance schedules are given for Monju [22] including parameter variations for drum boiler and once-through systems, and for Super Phenix at Creys Malville [27]. In the literature data is also available for Russian and German units [21,69].

5.2 Dynamic Characteristics

(i) PWR

FIGURE 13(b) illustrates the response characteristics of a typical 660 MWe PWR operating at 80% MCR to a step change in load demand of +100 MWe (~ 20%). The high power density coupled with high heat transfer characteristics for the NSSS results in a relatively fast response capability to load changes. The diagram shows that a plant manoeuvre is complete within about 60 seconds of an applied demand.

A further example which illustrates the rapid response rate of water reactors is shown in the response of a 1300MW PWR at Grafenrheinfeld (FRG) to a step demand of ~ + 10% (FIGURE 9). The diagrams show clearly on a short and longer timescale the rapid response available from such reactors. In this case the boiler stored energy is released upon operation of the steam valves, followed up by the actions of the control system (RFT/CC) which increases reactor power at about 30% m^{-1} almost immediately after the applied change. See also FIGURE 46.

(ii) BWR, SGHWR

FIGURE 14(b) illustrates the response characteristics of a typical BWR for step load demand of + 10%. Again, as with other water reactor systems, the response rate is fairly rapid. The responses shown in the example are those obtained from a commissioning test on a 660MW BWR operating in coordinated control, and TFR modes. (The Reactor power trace is in 'tonnes per hour, steamflow'.)

FIGURE 17 illustrates the response of an SGHWR to a step demand of + 10% in reactor power at ~ 80% load.

Tests which were carried out in 1977 on the SGHWR at Winfrith demonstrated a capability of ± 10% step changes in loads at various loads, ramp rates of ± 10% per minute between 40-100% load in TFR mode (also referred to as 'de-coupled' mode) and ± 7% per minute in a similar load range in RFT mode (also referred to as 'coupled' mode). The slightly lower rate for the RFT mode was necessary to alleviate excursions in steam pressure and drum water level, which in the case of direct cycle plant can incur trips of the reactor.

(iii) GCR

FIGURE 15(b) illustrates the dynamic characteristics of a 660 MWe AGR plant as obtained during a planned trip of the plant from 180 MWe to houseload, approximately 15 MW. The NSSS is fitted with 40% capacity HP turbine and 60% capacity IP + LP turbin bypass systems, and the diagram shows the operations of the HP and IP bypass systems. Tripping to Houseload was achieved successfully in this test. The use of high speed digital data recording is beneficial during such events to aid fault-finding - See also 6.1(iv). In comparison with water reactor systems such as shown in FIGURES 13 and 14, the NSSS settling times are seen to be longer, this being due to the greater thermal capacity of the system which includes a reheater within the NSSS coupled with poorer heat transfer rates overall for the gas-cooled system (in contrast to the water-cooled systems). Against this however, the temperature differentials and rates of change are alleviated so reducing plant stressing during load changes.

(iv) HWR

FIGURE 16(b) illustrates the response characteristic of a 850MWe CANDU HWR to a 10% MCR step increase in turbine power (near full load) due to a grid frequency decline. This step increase in turbine steam flow demand in the Reactor Following Turbine mode causes the reactor to follow the demand by utilizing the reactor power control to maintain constant boiler pressure. The power demand is sustained within one minute.

5.3 Examples of responses to electrical disturbances

The following examples illustrate the behaviour of typical nuclear plants to disturbances which can occur both within the NSSS, for example, loss of a coolant pump, and without, such as load rejection to houseload, or a grid disturbance.

(i) PWR

FIGURE 19(a) illustrates an automatic set-back of the reactor in the case of loss of one main coolant pump which is detected by a speed measurement. A load reduction to ~ 45% is caused by dropping sufficient control rods simultaneously into the core thus reducing reactor power very quickly (<3 sec). The turbogenerator is also set-back by the control system (CC) and the plant achieves a new steady state between 1-2 minutes [50].

FIGURE 19(b) shows the transients recorded during a load rejection from 80% to houseload with initiation of turbine bypass valves to condenser coupled with rod-pair dropping at 2s intervals so as to achieve the bypass capacity loading of ~ 50% MCR steam flow. A new steady state is achieved in about 2 minutes [50].

(ii) BWR

FIGURE 18 shows the power output and grid frequency traces for the SGHWR located at Winfrith Heath in Dorset, S. England at 0940h on 25 June 1974. In this incident there was a 550MW generation loss from the Scottish system brought about by the tripping of a unit at Longannet. Although the generating capacity of the SGHWR (UKAEA) is small by comparison, its generation contribution to grid support is clearly evident in the traces.* Because of the coolant flow control in the BWR with a direct cycle, its dynamic performance is in the same class as the PWR - perhaps even better in the upper power range.

* The Scottish System and UKAEA generation are interconnected with the CEGB system.

(iii) HWR

Most HWRs can operate in a Reactor Following Turbine (normal) or Turbine Following Reactor (alternate) mode of control [58].

The recordings in FIGURE 20 illustrate the response of a typical HWR-CANDU to a turbine load rejection and "trip to houseload" manoeuvre. This event initially causes a very fast reduction in steam flow; the controls initiate a reactor power "stepback" (rapid reactor power reduction to below the 60% level); and causes overall unit control mode to change from 'normal' (RFT) to 'alternate' (TFR). Boiler pressure is seen to increase rapidly and control is effected by the boiler pressure controller using the steam bypass Atmospheric Steam Dump Valves and Condenser Steam Dump Valves. The reactor power would be adjusted by the operator so that it would settle near its "stepback" endpoint of ~ 60% MCR. The unit continues to operate, supplying houseload and routing the excess steam to the condenser via the turbine bypass system in the "poison-prevent" mode.

(iv) FBR - Dounreay, Scotland

FIGURE 21 shows the response of the Dounreay 250 MW FBR to a scheduled load reduction from a part load condition to a lower load, initiated by a ramp reduction in primary and secondary Sodium Flows. Steam pressure was controlled manually equating the response to a TFR mode. The unit was designed to operate in an RFT mode. The load reduction was necessary to accommodate loss of a Circulating Water Pump.

5.4 Experience of nuclear plant performance in small grids

Information was provided by CNEA (Argentina), AECL (Canada), and KWU on the operation of HWR-based NSSS (KWU and CANDU) in small grids in Argentina (Embalse) and South Korea (Wolsung) which are subject to continuous frequency swings and excursions of the order of ± 1 Hz; load cycling and severe grid upsets.

A 357 MW HWR (KWU) has been in operation in Argentina since mid 1974, at which time the network had an installed capacity of 4,560 MW with frequency fluctuations amounting to up to 2 percent. The closed loop control normally operated according to a TFR mode (constant load requirement) albeit that RFT mode was selected in the event of high fluctuations (load-following requirement).

In addition, a 648 MW HWR (CANDU) was installed in mid 1983, at which time the network had a capacity of 8,330 MW. Although this gave rise to a considerable improvement in many respects, the network frequency control characteristic still remained poor exhibiting relatively common fluctuations of 0.5 percent (0.25 Hz) from peak to peak. The primary control of the CANDU unit was originally set at 5% droop under an RFT mode of Control, but this led to large swings in reactor power. The control was therefore set to TFR but more recently CNEA have again used RFT but included a deadband of ± 0.5% speed error and increased the governor droop to 8% to minimise power fluctuations in the reactor. Under light load conditions these units are often called upon to load-follow.

The 680 MW unit at Wolsung is part of a 5000 MW grid; frequency control is poor with typical short term fluctuations of about 0.5% (0.3 Hz) peak to peak lasting about a minute and occurring once every 20 minutes. Larger swings of up to ± 1 Hz occur less frequently. These frequency fluctuations cause reactor power changes at about ± 3.5%m^{-1}. To minimize these continuous fluctuations, the units are operated in the TFR mode and the load limiters are set close to the turbine load set points.

5.5 Behaviour of Major Auxiliaries during power supply disturbances

The behaviour of plant auxiliaries within a NSSS during voltage and frequency excursions are of importance in determining the dynamic behaviour characteristics, overall, of the NSSS. Continued operation, as for example during a successful TTH event, depends on the ability of all auxiliary systems, including the control systems, to provide the correct support function during severe transients. The dynamic behaviour of individual components will be governed by many factors such as unit load, inertial energy, circuit configuration - impedance levels, and the severity of the voltage and frequency induced effects. In turn, degradation of performance of auxiliaries may influence adversely the operating 'margins to trip' within the NSSS.

An example of this occurs in the PWR - based system, and the 'Departure from Nucleate Boiling (DNB)' margin - a reducing pump speed will increase the voidage local to the fuel elements, with a possibility of increased heat fluxes in parts of the core [40,48]. In a BWR this would not be detrimental for short-term transients.

The measurements of voltages, currents and frequency during plant commissioning and proving for major auxiliaries is recommended as an aid to understanding the implications of events such as TTH, partial or total load rejection etc. on auxiliaries, and on the NSSS. Such tests have been carried out on a twin-reactor station in the CEGB which is prone to 'islanding' - refer FIGURE 22. Most of the major auxiliaries on this NSSS are electrically driven including the boiler feed pumps and the results of plant tests provided useful information on the protection settings for these items of plant. FIGURE 23 shows the current/frequency characteristics, both steady state and dynamic for the major auxiliaries on the station, such as would occur following a load rejection (frequency rising). A further example of the onerous conditions prevailing on another twin-reactor station in the SE of England during a grid event can be seen from FIGURE 25. This represents a 'disturbed grid' situation (voltage and frequency falling). It will be noted that stable conditions were achieved following these events without trips being incurred on either reactor. A brief description of the events is given in Section 7.4.

The 'disturbed grid' event coupled with load rejection leading to islanding of a plant in a generation-deficient island presents the most onerous condition of operation. The specification for event and data recording systems for application to NSSSs should take account of likely power supply fluctuations which occur during 'disturbed-grid' and 'load rejection' events. Carefully monitored NSSS behaviour during disturbed-grid and separation events is a desirable objective but in practice data is usually sparse.

6. EXAMPLES OF TTH AND ISLANDING EVENTS

The following examples embrace the broad definition of 'islanding' - that is those events which are brought about by a sudden change in the consumer demand level (excluding changes in houseload) as a result of switching operations which separate the NSSS electrical system from the main grid. It includes those events which result in separation of the NSSS from part of the grid, creating small or large islands, and leaving a residual load on the NSSS which exceeds the houseload. When the residual load matches exactly the houseload the islanding event corrresponds to 'Tripping to Houseload'. An unsuccessful TTH event is defined here as one which leads to a reactor trip. TTH has been the subject of a special report by WG04 [35].

6.1 TTH Events

(i) BWRs : Data on TTH events which have occurred on 7 out of 8 BWRs currently in operation in Sweden shows that in the period 1972-1984, during tests, six TTH events were successful. A successful TTH event is regarded by the Swedish utilities as one which is initiated by a grid event which is followed by a period of operation exceeding 10m of the NSSS on houseload. In operation, with grid-induced disturbances three events were successful. Each of the plants is fitted with ~ 100% steam bypasses to the condenser. Experience gained from unsuccessful TTH events on Swedish NPPs shows that a frequent cause of unit trips, is the imperfect synchronization of IP interceptor valves (see also Example (iv) below).

(ii) HWRs - CANDU : Experience with tripping to houseload for 8 nuclear units in Ontario Hydro over 5 years shows that out of 19 potential TTH events 17 were achieved successfuly (90%). This is an improvement over fossil-fired units over a five year period for which the utility reports that only 10 out of 15 potential TTH events were achieved successfully (67%).

(iii) FBR : At the time of the incident the Fast Reactor at Dounreay was generating about 100MW when the grid connection was lost, leaving a residual load of 16MW (houseload). The NSSS is fitted with a 50% capacity steam bypass to the condenser [20], which is brought into service by a boiler pressure signal, but limited to operate from mid-range power output levels only. Thus, on this occasion, the bypass operated as expected and the new equilibrium on houseload was established in 20 seconds. The NSSS was islanded without a reactor trip, but the ensuing sustained high frequency, ~ 53Hz, contributed to the trip of a secondary circuit sodium pump about 7 minutes after the disconnection which lead to a reactor trip [13].

(iv) AGR : FIGURE 26 shows the result of a commissioning test carried out at Dungeness 'B' Power Station. In this test, the HV breaker was opened to simulate a TTH event. It will be seen from the traces presented that the reactor power was reduced by the control system by reducing the speed of the gas circulators. This is achieved by the change to RFT philosophy from a TFR strategy which is the normal operating mode. The results shown were obtained with the aid of a high speed digital recording system which is permanently available as a watchdog facility and post incident recorder. This system has identified areas for improvement, particularly in the timing and synchronization of HP and Interceptor Valve operation to the benefit and reliability of successful TTH events.

6.2 Islanding Events

(i) England : 25/26 April 1981

Severe blizzards with heavy falls of snow accompanied by high winds in SW England and S.Wales over the weekend 25/26 April 1981, led to unprecedented system conditions from late evening Saturday to midday Sunday. In a little over 12 hours, 350 supergrid and 132kV faults were experienced in the Bristol Grid Control Area (GCA) this being equivalent to the number normally experienced during a typical year. At the height of the snow storm, faults were occurring at about 90 second intervals, leading eventually to loss of 275kV lines in the Bristol Area.

One twin reactor Magnox Station was generating at full power exporting 437MW. From 2200 hrs on Saturday to 0330 hours on Sunday the station was subjected to severe voltage and load swings leading eventually to a load regime of c.140MW total export by about 0600 hrs. Following an instruction from Grid Control this load was maintained in anticipation of further transmission system losses with the possibility of the station being required to supply the load in the Bristol area via a 132kV line. The transmission system continued to suffer faults until 1114 hrs when the station became detached from the grid and supplied the Bristol area. The operators adjusted the two reactors to balance the load and controlled the turbine speeds manually until 11.59 hrs when reconnection to the grid was established. The frequency records show severe excursions with extreme values of + 51.05Hz and 49.0Hz. The voltage swings caused several items of non-critical auxiliary plant to trip which were not equipped with latch-in contactors. The generation was re-established to 438 MW by 2148 hrs.

(ii) England : 13 December 1981

Severe blizzards accompanied by high winds and snowdrifts affected the NW, Wales, South West and Southern England on 13 December 1981.

Two incidents which occurred in the North West led to islanding and sustained operation of a 1000 MW nuclear power station for nearly six hours pending its reconnection to the main grid [37]. The operators reduced gas flows on both reactors to minimize steam loss and controlled the turbine speeds manually. FIGURE 22 shows the frequency records and loading information during these events. Successful islanded operation was achieved by the correct operation of control systems aided by operations staff skilled in dealing with such incidents - such skills having been developed by experience of previous incidents, evolutionary developments in station operating rules, and a good understanding by the operators of the plant dynamic characteristics. The plant in question has a configuration as shown in FIGURE 4(c) with no provision made for live-steam bypassing of the turbine to the condenser. Normal control mode is TFR, so that in a load rejection circumstance the operator requires to set the steam pressure control to 'Manual' as part of the Station Operating Instructions. Depending on the likelihood of restoring the grid connection (and the duration of separation) the operator decides whether to reduce load so as to minimise the loss of steam and feedwater. Reliance is placed on steam emission through the boiler safety valves to cope with the excess steam generation in a transient situation, and adequate reserves of feedwater. The effect of overfrequency established during tests of similar incidents is shown in FIGURE 23.

The blizzards and high winds also affected the SW peninsula. In the Bristol Grid Control Area, 407 supergrid and 132 KV faults were recorded between 0930 and 1800 hrs. Most of these were caused by high winds inducing movements and clashes of ice laden overhead lines. It was estimated that 1700MW of demand were lost in the Bristol Area at the peak of the storm. Temperatures were recorded at -20°C in exposed areas. An AGR station with two operating reactors, delivering 460 MW, a Magnox Station with two operating reactors, delivering 440 MW, and an SGHWR delivering about 90MW were connected to the system at the time of the events.

Following a transmission fault, the operation of over frequency protection relays on one of the AGR's gas circulators brought about a reactor trip. Loss of this generation caused voltage and frequency dips which tripped the other AGR gas circulators on underfrequency tripping that reactor. Following further disturbances at 1303 hrs the line connecting the SW peninsula to the main grid was lost leaving the Magnox Station islanded with a generation deficiency of ~ 360 MW. This led to a large voltage depression followed by a rapid drop in frequency. Low frequency relays disconnected ~ 200 MW, and Grid Control operator action shed a further 200 MW, enabling the Magnox Station to float the balance. Unfortunately, the reactor control rod system reverted to 'DC - hold' as a result of transients from the islanding. Normal frequency was not recovered and fell steadily until about 1318 hrs when grid conditions deteriorated rapidly and the plant was tripped by the operator at 1319 hrs. Thus islanding and continued operation of the Magnox Station could not be sustained beyond 16 minutes on this occasion.

The 400KV grid system which connects with the 132KV system in the vicinity of the Winfrith SGHWR experienced several faults in quick succession [13]. The 400/132KV connection was lost during the storms leaving the Winfrith plant islanded via the 132KV local system, attempting to supply a load estimated to be about 200 MW. At the time of the incident the plant was generating 90 MWso (Full Load = 100 MWso). This caused a voltage decrease and the automatic voltage regulator increased generator excitation in an attempt to meet the additional grid demand. This was unsuccessful as there appeared to be no additional generation in the 'island', and the voltage levels fell rapidly to 8.4kV (13.8 KV) and 84.3KV (132 KV) at the generator and grid busbars respectively, with a supply frequency of 46.9Hz (~ 2800 rpm). The grid transformer voltage fell from 11KV to 6.6KV. The voltage transients following the islanding precipitated a trip of a primary coolant pump leading to a reactor trip 1.5 seconds after loss of the 400/132KV connection. This sequence corresponds to a 'disturbed grid' event coupled with a grid disconnection and 'islanding'. Supplies were restored via the 400/132KV connection 38 minutes after the initiating event but the reactor was not returned to service for 30 hours. The delay was due to blizzard conditions disrupting cooling water supplies.

(iii) Ontario, Canada : April 1972 (HWR-CANDU)

In an electrical disturbance in Ontario in April 1974, Canada, two CANDU nuclear units (loaded to 500 MW each) were contained in a large electrical island which was subject to over frequency of 62.5 Hz. One of the units was tripped manually following generation oscillations of between 40 and 300 MW due to erratic behaviour of the IP interceptor valves. It was resynchronized in 20 minutes and reloaded. The other unit continued in operation through the disturbance [38].

6.3 Lessons Learnt

From the limited data available, the successful outcome of TTH and islanding events is governed by the correct operation and timing of relevant functions. Valve operations, their timing and synchronism, and the behaviour of control systems (supported where necessary by operator action) in the presence of voltage and frequency disturbances are key factors which determine the outcome of grid disconnection events. Special provisions for high speed data capture and playback, preferably using digital recording techniques can provide a useful aid to the determination of causative factors and the sequence of events.

7. BEHAVIOUR OF NUCLEAR PLANT DURING MAJOR GRID EVENTS

An EPRI Study of Nuclear Plant Response to Grid Electrical Disturbances was reported in 1983 [40]. It considers a number of incidents in the USA and the impact of them on nuclear plant operating at the time. This paper examines European incidents of grid disturbance and their effect on Nuclear Power Plants.

Disturbances to grid operation have occurred in Sweden, Belgium and France and these have been the subject of reports [65] [59] [60]. FIGURE 24 shows the reconnection and reloading histories for these events. Of relevance to this paper is the dynamic behaviour of Nuclear Power Stations during such events. Of the four recorded incidents depicted in the Figure, three are chosen for analysis below. In all cases system recovery and reconnection of generation to meet demand was complete between 3 and 8 hours, but the reinstatement of the NPPs, took much longer in the French and Swedish incidents due to poisoning-out of the reactors.

7.1 Swedish incident : 27 December 1983 [65]

At 12h57 on 27 December 1983, the Swedish power system was subjected to a disturbance initiated by failure of an isolator in a 400/220 kV substation located about 50 km north-west of Stockholm. The incident led to the loss of two out of seven 400 kV north-south ties, with consequent overloading and subsequent tripping out of the remaining ties within one minute. Prior to the event, the total power generated/supplied was 18300 MW (including imports) with a net export north to south of 5600 MW. Hydro plant in the north accounted for about 11000 MW, with the balance being made up in the south by 5800 MW nuclear and fossil, and imports of about 1500 MW.

Within a minute of system separation, excess generation of 6000 MW in the northern system resulted in a frequency increase to 54 Hz in that area in 5 seconds with a fallback to 50 Hz within 15 seconds following tripping of some hydro plants by overfrequency protection circuits. Similarly, in the southern system there occurred a generation deficiency of about 7000 MW and the frequency fell rapidly at 2-4 Hz sec^{-1}, as did the voltage. It was noted subsequently that only about half of the load-shedding relays actually operated on low frequency; the remainder tripped on loss of voltage.

The ten nuclear power plants in Sweden are all situated in the central and southern regions, and normally operate at base load. During the incident eight were operating, 7 BWRs and 1 PWR. All these plants have provision for Tripping To Houseload and 100% steam dumping capacity, and have been demonstrated to accommodate load rejection from Full Load to House Load. However, the nature of this disturbance did not give the nuclear stations a good chance to continue in operation. Except for Forsmark 1 which was connected to the northern system, all of them shut down. They coped successfully in terms of primary control but did not contribute to the restoration of generation supply during the first hours following the disturbance. In this incident, the disturbance to the grid was so violent that not only the HV breaker on the grid-side but also the LV breaker on the generator-side opened so that the basic conditions for Houseload operation could not be satisfied.

Most of the nuclear power plants in Sweden are equipped with two gas turbines. All of these started up automatically except for one at Oskarshamn and one at Barsebäck, both of which required manual intervention to initiate start and synchronization. Within +1h of the disturbance the 400 and 220 kV grids were intact and reconnections established to consumers in the south. Full scale supplies were restored within +5½h. The first nuclear unit to resynchronize and load was Ringhals 2 at +21h35 and the last was Oskarshamn 1 at +50h approximately.

7.2 Belgian incident : 4 August 1982 [59]

In the summer, the grid loading is light, and at 11.00 hrs when the incident occurred was ~ 5500 MW, that is about 45% of installed capacity and 70% of peak demand. At the time there was a small import of 3% active power and export of 4% reactive power to neighbouring countries interconnected with the Belgian system. Just prior to the incident the Belgian generation/load profile consisted of:

GENERATION			DEMAND	SPINNING RESERVE
Nuclear (%)		Hydro + Fossil + Export (%)	TOTALS	Fossil %
MW	~ 2400 (44)	3100 (56)	5500	1300 (23)
MVar	~ 900 (36)	1600 (64)	2500	800 (32)

The nuclear component comprised 4 PWRs, three at Doel and one at Tihange, and with the Hydro input from Coo Station of 1000 MW, accounted for 62% of the demand. These units were connected to the 380 kV network, and the balance of load and spinning reserve was supplied by the other plants connected to the 150, 70 and 36 kV systems, all interconnected to the supergrid. The load was largely inductive, because of a high percentage of industrial load comprising mainly motors.

The event was triggered by a turbine trip initiated by a turbine protection device on Doel No.3 unit with a consequent generation loss of 700 MW and 450 MVars. The ensuing voltage drop on the 380 kV grid initiated a boost to the excitation on the adjacent Units 1 and 2 at Doel via the AVRs, which soon reached the upper excitation current limit. Protection arrangements intervened to reduce the excitation levels but AVR action raised the currents again leading to unstable behaviour. As a result of loss of reactive power generation the voltage levels continued to fall. Within 4½ minutes of the Doel 3 trip the protection relays successfully tripped 5 turbo-generators to avoid overheating in the machines due to overload - Schelle 31 and 32, Pont Brûlé 3 and Doel 1 and 2. The 125 MW Pont Brûlé turbo-generator was by then totally destroyed. These trips reduced generation by a further 1800 MW and 1100 MVar in the Northern and Central parts of the system leading to a x2 rated value of current in the 380 kV South to North tie at a voltage level of about 75%, coupled with voltage and power oscillations between North and South. Within seconds this link then tripped.

These generation deficits in the North and Centre could not be compensated by the available spinning reserve via the 150/70/36kV subsystems.

Cascade trippings occurred which divided the Northern section of the grid into 3 zones - a Brussels/Antwerp island, a Western zone, and the Limburg zone which remained connected to the Southern zone by a 150 kV line at low voltage. Trips in the 150/70 kV system led to a loss of 2400 MW, i.e. 44% of the loading prior to the incident.

The summary report of this incident [59] does not give specific information on the thermal response of the NSSS for Doel 1-3 and Tihange 1 but no adverse effects were recorded at the nuclear stations, and safe shut downs resulted. As can be seen from the survey (Part 2), the Belgian PWR plants have generous provision of Steam Bypass capacity to the condenser, but, clearly, in the presence of protection-initiated shutdown of the turbo-generators at Doel there was no scope, because of the gravity of the incident, to warrant continued running at low power when there were deficiencies in active and particularly reactive power generation. It was the latter that led to a worsening situation.

Of the lessons learnt as a result of this event, the generator protection has been modified to delay action by 20 seconds in the event of excitation current levels exceeding the rated value. After this period the current level is reduced to 70-80% rated value. In the absence of effective control action the generator trip is initiated after 30 seconds. To reduce vulnerability to major grid events, other actions have been taken to reinforce the grid and distribute the generation pattern more securely in terms of both active and reactive power.

7.3 French incident : 19 December 1978 [60]

The blackout which affected a large part of the French network on 19 December 1978 was initiated by an overload of the 400 kV Bezaumont-Creney Line. It was followed by total interruption in supplies to domestic and industrial users with the exception of the North, East, South-East and Alpine Regions. The maximum loss of load was ~ 28000 MW on a demand of ~ 38000 MW, i.e. a loss of about 75%.

A widespread regime of instability, characterized by oscillations of voltage and current, affected the whole of the French network except for the Alsace, Lorraine and Ardennes regions which remained coupled to the European continental network. In order to limit the consequences of severe electrical disturbances, the current safety philosophy is based on:

(i) the creation of separate zones of the network activated by low frequency and also by frequency instability, and

(ii) automatic load-shedding at low frequencies

At the time of the incident, 14 nuclear units were connected to the network, and of these 11 were in the regions affected and comprised 8 GCRs, 1 HWR, 1 PWR, and 1 FBR. The impact of the grid events on these nuclear plants was as follows:

- 1 successful isolation, i.e. 'islanding'
- 3 attempted but unsuccessful isolations
- 4 plants tripped via internal protection systems
- 1 manual trip without attempting isolation
- 2 plants tripped via internal protection systems (isolated operation not possible by design)

The loss of supplies to consumers varied from 13 minutes to a few hours, with an average of ½ an hour. All the internal supplies fed from diesels and other auxiliary groups functioned correctly, and the safety of the plant was assured during the events. Most of the NPPs were reconnected to the system within a period of 24-36 hours. The delay is explained by the fact that most of these plants were fuelled by natural uranium and the build up of Xenon following trips from full power caused poisoning out for 24 hours. The PWR unit was reconnected 13 hours after reinstatement of supplies.

7.4 British incident : 28 November 1980

A supergrid circuit fault and subsequent circuit trippings caused a system disturbance in SE England which lasted for 19 minutes. A general depression of grid system voltage occurred and a twin reactor Magnox Station was left connected via a high impedance path to the main network resulting in voltage and frequency oscillations. Islanding itself did not occur, but the fluctuations in power supplies led to the automatic disconnection of some consumers. Although the scale of this incident is not as great as the previous events, the consequences of it on the electrical plant (being fully recorded) will be of direct interest. During the course of this incident, the rotors of the turbo-generators oscillated with respect to the system. Severe voltage frequency, MW and MVAr oscillations were monitored during the disturbance and a sample of the results are presented in FIGURE 25. There was a loss of generation of about 100 MW in the course of the incident. The operators coped with numerous alarms and with the tripping of control rod systems, but throughout the events the reactors remained operationally very stable. All other control systems responded correctly, and no station auxiliaries were lost.

8. DEVELOPMENT AND REDESIGN OF EXISTING CONTROL SYSTEMS FOR NPPs.

Some of the early power reactors which are still in service after 20 years or more of operation will contain control and instrumentation equipment which is obsolescent and which cannot be repaired or maintained economically. Thus, the matter of equipment refurbishment is an ever-present consideration by utilities.

8.1 Considerations arising from changes in the operating regime.

It may occur that the operating regime of a nuclear plant may not correspond to the regime for which the original control scheme was specified. An example is the use of a base-load plant for regulating duty which may arise from circumstances beyond the control of the utility. The load cycling of nuclear plant in the future is clearly an evolution which will bring about changes in operating regime. Alterations to grid connections, and variation to philosophy on auxiliary and guaranteed supplies may also influence specifications and operating regimes.

Utilities need to keep under review the harmonization of operating requirements and control system objectives, since evolutionary changes in the former may not be possible without changes to the latter. Whilst this will not affect safety considerations, it may affect reliability and quality of control.

8.2 Considerations arising from the need to refurbish ageing Control Systems

With the adoption of distributed computer control and ready availability of micro processors at an economic cost, the power and scope of control systems, including achievement of a high degree of flexibility, reliability, and self-checking enables very sophisticated systems to be devised. However, the price to be paid for such elegance in terms of hardware can be misleading in economic analysis if no account is taken of the costs of software development (including its QA route) and proving [67]. Just as the hardware will require replacement in due course, so too will the software. It is therefore of great importance for utilities to establish clearly the operating requirements, and to set these down as a baseline of needs. For stations with an operating history and a good estimate of future operational requirements, the documentation of needs is an essential precursor to a refurbishment policy. This is especially so where software design is concerned since the basic overall project costs may be significantly influenced by it.

9. STATE OF THE ART AND PROSPECTS FOR THE FUTURE

This review of international practices and of objectives in design targets and operating regimes for NSSSs and their control systems in support of system control, and experiences of actual behaviour of nuclear power stations during major grid disturbances and incidents leads to the following observations:

(a) World Scenario

9.1 Growth - The growth of nuclear capacity is continuing but at a lower rate than forecast a decade ago. The forecast for 1990 of 370 GWso from 509 NPPs in operation and construction in the World is an increase over 1985 of 97 GWso or 36% capacity from 405 reactors [2] [9.2)].

(b) This Survey of NSSS

9.2 Scope - Coverage in this survey which includes a substantial world sample examines 203 GW of operational nuclear capacity from 290 reactors (1985 basis) that is 93% of world capacity from 344 reactors operating at end 1984. The full survey includes 282 GWso of nuclear capacity from 373 reactors (1990 basis) or 76% of world capacity (operating now and in construction - FIGURES 27-29.

9.3 Reactor Types - Water reactors dominate in choice of NSSS by utilities. Within the total family of 373 reactors which are operating now and/or expected to be in service by 1990, ~ 85% are water-cooled reactors, and they contribute ~ 93% of the total nuclear capacity of 282 GWso. The greatest proportion of the capacity is contributed by the PWR (~ 61%) followed by the BWR (~ 19%). Including the Russian LWGR in the BWR category increases the figure for BWRs to 26%. The balance is made up of HWRs (6%) GCRs (~ 6%). Balance, FBR (1%) - FIGURES 27,28.

9.4 Nuclear planting - The growth rates for the period to 1990 are greatest for France, Belgium, Japan, FRG, USA and the USSR, whilst the proportion of nuclear capacity to the system capacity > 20% are forecast for Belgium, France, Germany, Finland, Korea, Taiwan and Sweden, with the highest percentages applying to Belgium (~ 35%) and France (~ 53%).

The percentage nuclear component of system capacities for the countries adopting nuclear power is showing a continuing rise in the current decade comparable with the previous decade. - TABLE 1 and FIGURE 2.

9.5 Turbine Governing -

(i) Most plants surveyed were governed in the HP mode, but where IP steam admission valves were provided (nearly all the sample) IP governing was in most cases also provided. The replies have not been included in the database.

(ii) Primary Control - The regulating band gives an indication of the extent of 'base-load' required for nuclear plant. The survey shows that of 373 reactors less than ½% exercise >10%, 16% exercise between 6 and 10%, 5% exercise ≤ 5%, and ~ 50% do not participate in primary control. The balance of ~ 30% were 'unknown' (includes non-operational plant). - FIGURE 30.

(iii) Average speed droop - For recently commissioned and plant in construction an adjustment to governor sensitivity is available to the operator, whereas for earlier plant no adjustment is possible with some exceptions [49]. The average value of droop lies between 2.5 and 5.0% for ~ 83% of the plant in a sample of 81% of the overall. FIGURE 31.

(iv) Average Deadband - Most of the NPPs examined showed that most plants operate without any frequency deadband. Excluding those replies where no deadband is stated, only 17% of the sample operated with deadbands between ± 50mHz and ± 500mHz - FIGURE 32.

(v) Normal setting of valve limit - Nearly all nuclear units operate to fully open steam valves.

9.6 Secondary Control -

(i) Provision for secondary control is made for 91% of the reactors in an overall sample of 98% of the survey - FIGURE 33.

(ii) Secondary control mode, i.e. RFT, TFR, CC for a 90% sample of the overall shows that RFT is favoured for nearly half the sample (48%) with smaller groupings for CC and TFR. Some reactors, ~ 25%, have more than one mode. - FIGURE 34.

(iii) Utilization of secondary control - Most reactors in operation now (1985 basis) do not contribute to secondary load control. In the sample of ~ 77% of the surveyed reactors, only 6% participate in secondary control now i.e. the great majority are base-loaded. - FIGURE 35.

With regard to 1990, there is a significant shift indicated of operating regimes from base-load to load-fequency control. Within the sample of 72% of the survey, just under 50% expect to load-follow, with just over 50% not expecting to contribute in this way i.e. remaining as base-load units -FIGURE 36.

9.7 Control Range -

(i) Low load capability: In ~ 96% of the sample surveyed, nearly 60% of the reactors have a low-load operational capability of 20% or less. The balance have somewhat higher values of between 30 and 70% - FIGURE 37.

In terms of range and minimum load capability, nuclear power plants compare favourably with fossil-fired plant (oil and gas) and have a wider control range than most coal-fired plant.

(ii) Loading-rate: The normal loading rate for 60% of the sample (~ 95% of the total) was 4-5% MCR per minute with a small percentage of plant at the lower and upper ends of the spectrum i.e. 1-2% and 9-10% MCR min^{-1}. In general, nuclear plant is capable of comparable response rates to fossil-fuelled plant but in the case of water reactors the response rate is by comparison to other types much faster; small changes of about ± 5-10% MCR being achievable within 5-10 seconds from initiation. Commissioning tests on PWRs indicate higher loading rates viz. up to 60% per min in certain regimes of load - FIGURES 38, 9 and 10.

(iii)Power response characteristics - The response characteristics of the water reactors are faster than gas-cooled reactors. Only limited data are available on the response of FBRs, but all nuclear plants have a demonstrated capability for load-follow and frequency control duties. Typical dynamic response characteristics of NSSS of various types are given in Part 1 -Section 5.2 - FIGURES 9, 13-17.

(iv) Daily load-following capability - In view of the great majority of nuclear plant being predominantly base-loaded at present, see 9.6 (iii)) the response to this question yields the same results as for 'Control Range' - FIGURE 37. However, in view of the increasing role for nuclear plant in load and frequency control in the 1990s as indicated in 9.6 (iii), this topic is to be the subject of the follow-up study. [Refer Part 2 : Section 4.3 (b) and (e)].

For some countries with a high proportion of nuclear plant e.g. France, load-following by nuclear plant is in operation now.

9.8 Provision of Steam Bypasses to the Condenser -

Within a sample of ~ 97% of the survey, nearly all plants have some provisions for steam-bypassing of the turbine to the condenser. Over 50% have capacities in the range 80-105% of MCR steam flow (load) and 30% have capacities between 10 and 40%. - FIGURE 39.

Absence of such provisions in the design may not influence adversely operational flexibility except where local regulations do not permit steam venting to atmosphere via safety valves for extended periods.

9.9 Houseload - Half the reactors have a houseload of about 5%, with about 25% below and ~ 20% above it. This would be representative of the spectrum of residual load following a load rejection on such plant (excluding enhancement due to frequency rise effects or reductions due to the unloading of auxiliaries). - FIGURE 40.

9.10 Isolated operation -

(i) Capability: Within a sample of ~ 97% of the survey, only about 15% have no ability to operate in isolation to the grid. This implies a philosophy of reactor trip following a grid disconnection. The remainder of the plant, ~ 85% would expect to continue in operation following a grid disconnection (excluding spurious reactor trips) - FIGURE 41.

(ii) The review of Tripping to Houseload events (TTH) and major grid incidents shows that nuclear plant has performed well and with safety.

(iii)TTH - the capability for TTH in a survey sample of ~ 97% indicates a similar trend as for (i) in this group, about 17% would be expected to shut down on load rejection. The remainder have a capability of sustained operation on houseload. - FIGURE 42.

(iv) Residual Load - the spread of values given in the survey sample of ~ 98% indicates a strong bias corresponding to the houseload figures given in 9.9. This implies that the extensive provision of steam bypasses to the condenser, or reserves of feedwater to support atmospheric venting via safety valves enables operating regimes with very low load to be achieved by many plants. - FIGURE 43.

(v) Time period - For the same sample of plant, ~ 98%, nearly 80% can continue generating on houseload alone for ½ hour or more following grid disconnection - FIGURE 44. (Refer also Part 2 : Section 4.4(e)).

9.11 Review of NSSS behaviour during major grid events.

The examples given for grid incidents involving islanding, or disturbed grid events, show that nuclear power stations have operated correctly. In some cases they have continued in operation under onerous conditions pending reconnection, in others they have shut down safely as a result of protection operating or else manually by arrangement with grid control.

9.12 The low frequency operating and trip limits recommended by some manufacturers for turbines may be unduly conservative.

9.13 The understanding of unusual phenomena of rare ocurrence which are known to affect grids and NSSS large and small is a desirable objective. In the absence of such information it may not be possible to focus upon the key factors affecting subsequent behaviour. The provision of special monitoring equipment based on high speed information-capture using flexible digital recording (for example - refer FIGURE 26) aids the understanding of events to the advantage of future designs and improvement of safety and reliability.

9.14 The full compilation of replies to the questionnaire are included as Appendices A1-4 of this document (Part 2). The tabulations enable comparisons to be made for the NSSS for the full sample of 373 reactors operating in the 21 countries chosen and displayed in TABLES 1 and A1. It is proposed to retain the database for future use and studies by the Working Group.

The Database will provide an accurate record for the plants listed to 1990, but will not reflect changes brought about by plant retirement in the period 1985-1990.

9.15 Where redesign or development of control systems for existing nuclear power plants is being considered, utilities should review the operating requirements of the plant and provide a user-specification of needs. This will ensure harmonization of control system objectives and operating needs.

10. ACKNOWLEDGEMENTS - Refer Part 2 : Section 5.

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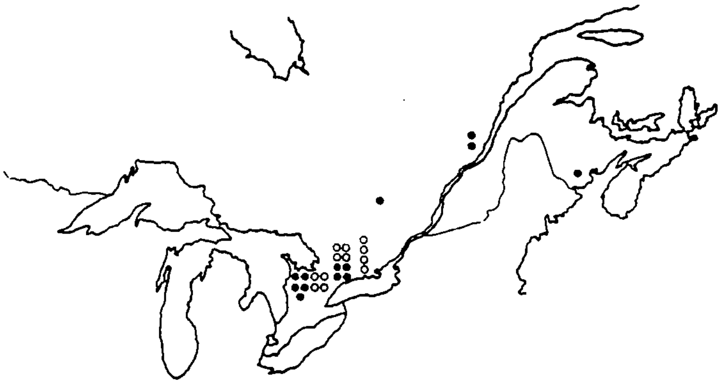
12. ABBREVIATIONS

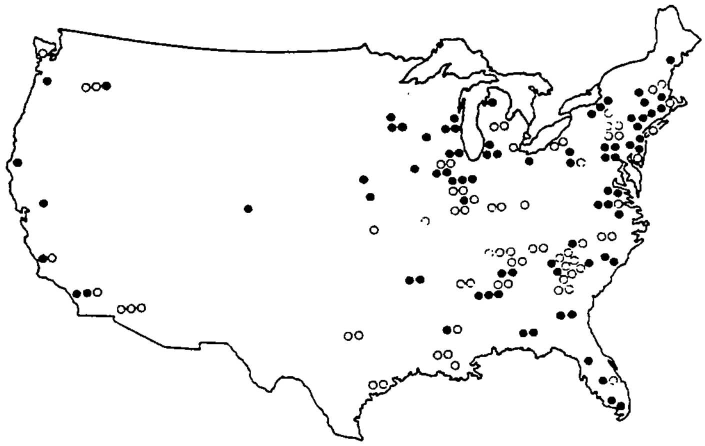
AGR	Advanced Gas Cooled Reactor
BWR	Boiling Water Reactor
CANDU	CANadian Deterium Uranium Oxide Reactor
CC	Co-ordinated Controls
CPV	Concrete Pressure Vessel
FBR	Fast Breeder Reactor
FTR	Frequency Trend Relay
GCR	Gas Cooled Reactor
GWe	Gigawatts electrical
GWso	Gigawatts sent out
HP	High Pressure
HTGR	High Temperature Gas Cooled Reactor
HWR	Heavy Water Reactor
Hz	Hertz
IP	Intermediate Pressure
LP	Low Pressure
LR	Load Rejection
LSR	Load Shedding Relay
LWGR	Light Water Graphite Reactor
MCR	Maximum Continuous Rating
mHz	Hertz x 10^{-3}
MWso	Megawatts sent out
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
PT	Pressure Tube
PV	Pressure Vessel
PWR	Pressurized Water Reactor
RBMK	A Russian Light Water Reactor (LWGR - Uranium-graphite Channel-type)
RFT	Reactor Following Turbine
RPM	Revolutions per minute
SGHWR	Steam Generating Heavy Water Reactor
SPV	Steel Pressure Vessel
SS	Start-Up/Shut Down
TFR	Turbine Following Reactor
TTH	Tripping to Houseload
VVER	A Russian PWR

Abbreviations for utilities as listed in Part 2 : Section 2 and in the Appendix - A1 : refer Reference [2].



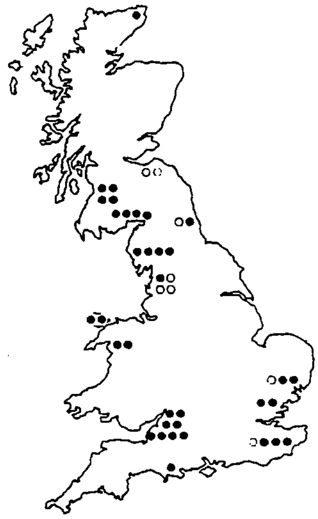
● Operating (Jan 83)	13 reactors	total capacity 6.5 GWe
○ Under construction or planned	12 reactors	total capacity 8.2 GWe

Nuclear power plants in Canada



● Operating (Jan 83)	81 reactors	total capacity 63 GWe
○ Under construction or planned	77 reactors	total capacity 77 GWe

Nuclear power plants in America

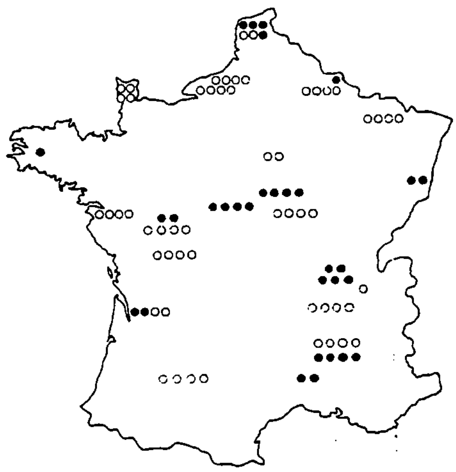


● Operating (Jan 83)	35 reactors	total capacity 13..5 GWe
○ Under construction or planned	8 reactors	total capacity 5..5 GWe

Nuclear power plants in Britain

ARGENTINA		
● Operating (Jan 83)	1 reactor	total capacity 0.3 GWe
○ Under construction or planned	2 reactors	total capacity 1.2 GWe
BRAZIL		
● Operating (Jan 83)	1 reactor	total capacity 0.6 GWe
○ Under construction or planned	2 reactors	total capacity 2.5 GWe
CUBA		
○ Under construction or planned	1 reactor	total capacity 0.4 GWe
EGYPT		
○ Under construction or planned	4 reactors	total capacity 1.8 GWe
INDIA		
● Operating (Jan 83)	4 reactors	total capacity 0.8 GWe
○ Under contruction or planned	6 reactors	total capacity 1.2 GWe
KOREA		
● Operating (Jan 83)	2 reactors	total capacity 1.2 GWe
○ Under construction or planned	7 reactors	total capacity 5.7 GWe
MEXICO		
○ Under construction or planned	2 reactors	total capacity 1.2 GWe
PAKISTAN		
● Operating (Jan 83)	1 reactor	total capacity 0.1 GWe
○ Under construction or planned	1 reactor	total capacity 0.6 GWe
PHILLIPINES		
○ Under construction or planned	1 reactor	total capacity 0.6 GWe
SOUTH AFRICA		
○ Under construction or planned	2 reactors	total capacity 1.8 GWe
TAIWAN		
● Operating (Jan 83)	4 reactors	total capacity 3 GWe
○ Under construction or planned	2 reactors	total capacity 1.8 GWe

Nuclear power plants in the developing world



● Operating (Jan 83)	31 reactors	total capacity 22.5 GWe
○ Under construction or planned	27 reactors	total capacity 30 GWe

Nuclear power plants in France

MAPS: Courtesy Simon Rippon 'Nuclear Energy

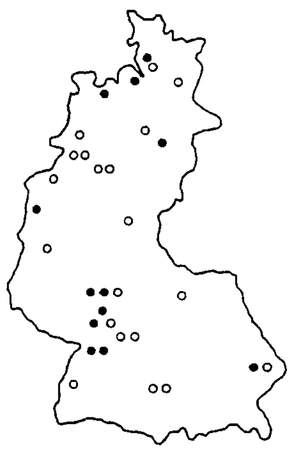
NUCLEAR

BELGIUM		
• Operating (Jan 83)	5 reactors	total capacity 3.6 GWe
o Under construction or planned	2 reactors	total capacity 1.9 GWe
FINLAND		
• Operating (Jan 83)	4 reactors	total capacity 2.2 GWe
ITALY		
• Operating (Jan 83)	3 reactors	total capacity 1.3 GWe
o Under construction or planned	9 reactors	total capacity 8 GWe
NETHERLANDS		
• Operating (Jan 83)	2 reactors	total capacity 0.5 GWe
SPAIN		
• Operating (Jan 83)	5 reactors	total capacity 3 GWe
o Under contruction or planned	11 reactors	total capacity 3 GWe
SWEDEN		
• Operating (Jan 83)	10 reactors	total capacity 7.7 GWe
o Under construction or planned	2 reactors	total capacity 2 GWe
SWITZERLAND		
• Operating (Jan 83)	4 reactors	total capacity 2 GWe
o Under construction or planned	2 reactors	total capacity 2 GWe

Nuclear power plants in Western Europe

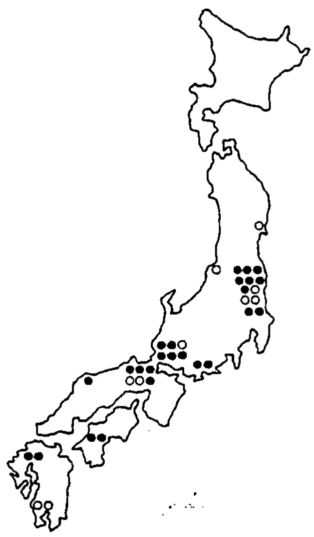
SOVIET UNION		
• Operating (Jan 83)	36 reactors	total capacity 17 GWe
o Under construction or planned	63 reactors	total capacity 63 GWe
BULGARIA		
• Operating (Jan 83)	3 reactors	total capacity 1.3 GWe
o Under construction or planned	1 reactor	total capacity 0.4 GWe
CZECHOSLOVAKIA		
• Operating (Jan 83)	2 reactors	total capacity 0.8 GWe
o Under construction or planned	15 reactors	total capacity 10 GWe
EAST GERMANY		
• Operating (Jan 83)	5 reactors	total capacity 2 GWe
o Under construction or planned	10 reactors	total capacity 4.4 GWe
HUNGARY		
o Under contruction or planned	4 reactors	total capacity 1.8 GWe
POLAND		
o Under construction or planned	4 reactors	total capacity 1.8 GWe
RUMANIA		
o Under construction or planned	4 reactors	total capacity 2.3 GWe
YUGOSLAVIA		
• Operating (Jan 83)	1 reactor	total capacity 0.6 GWe

Nuclear power plants in Eastern Europe and the Soviet Union



• Operating (Jan 83)	12 reactors	total capacity 10 GWe
o Under construction or planned	20 reactors	total capacity 24 GWe

Nuclear power plants in West Germany



• Operating (Jan 83)	25 reactors	total capacity 17 GWe
o Under construction or planned	10 reactors	total capacity 8 GWe

Nuclear power plants in Japan

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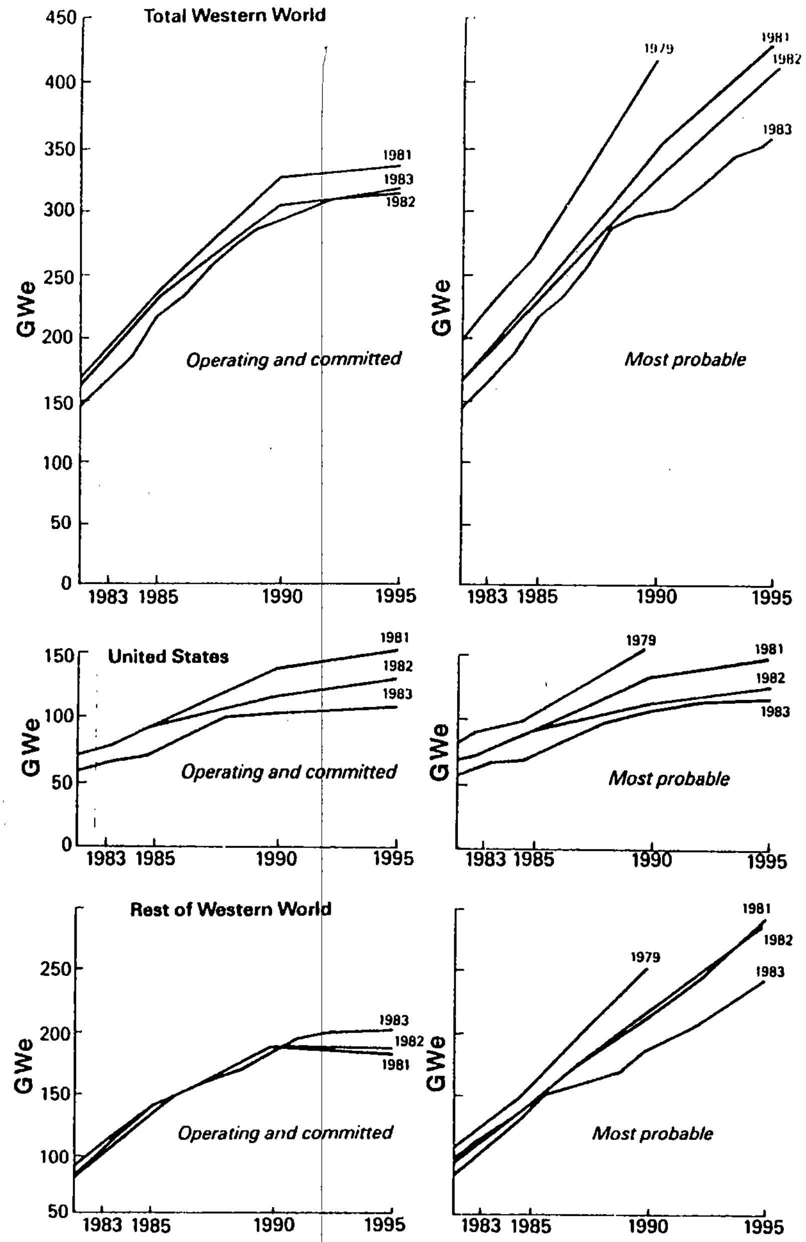
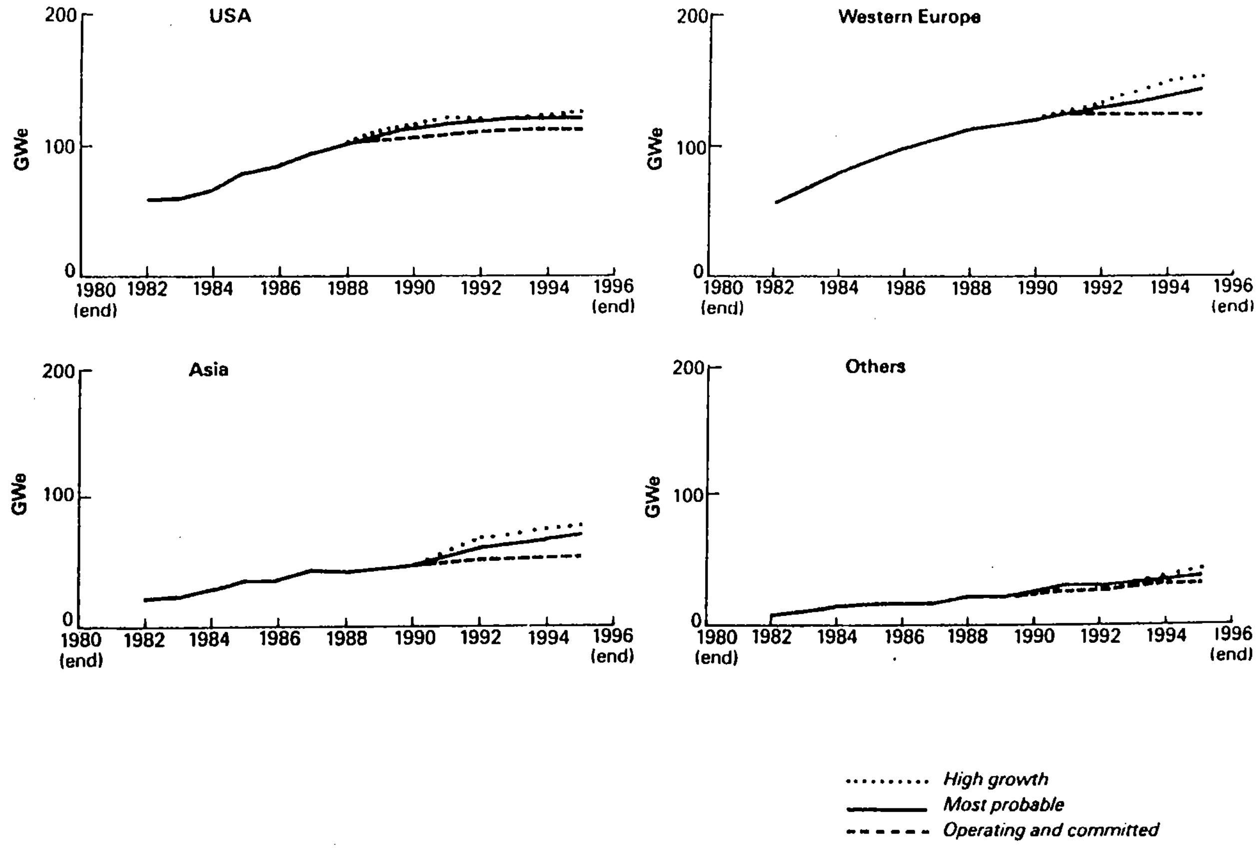
OWER PLANTS WORLDWIDE

FIGURE O

CIGRE SC39-04

Nuclear Plant Control





COMPARISON OF FORECASTS OF NUCLEAR GENERATING CAPACITY FOR THE WESTERN WORLD

FIGURE I (Courtesy – The Uranium Institute [4])

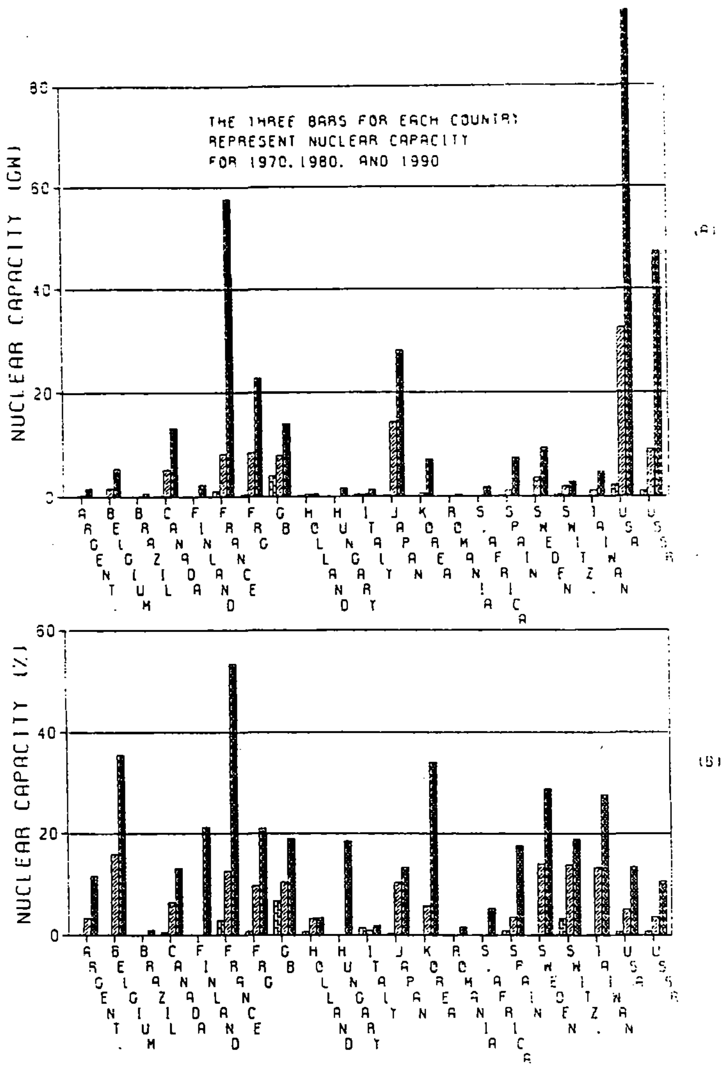


FIGURE 2 GROWTH OF NUCLEAR CAPACITY

	COMMISSIONED NUCLEAR CAPACITY GW50 AT START OF YEAR			SYSTEM CAPACITY GW60 AT START OF YEAR			NUCLEAR CAPACITY IN % OF SYSTEM AT START OF YEAR		
COUNTRY	1970	1980	1990	1970	1980	1990	1970	1980	1990
RA ARGENTINA	0	0.345	1.637	4.864	10.08	14.05	0	3.4	11.7
B BELGIUM	0	1.654	5.465	6.800	10.40	15.40	0	15.9	35.5
BR BRAZIL	0	0	0.626	10.35	30.28	58.46	0	0	1.1
CDN CANADA	0.206	5.266	13.29	42.80	81.60	100.4	0.5	6.5	13.2
SF FINLAND	0	I.I.	2.310	3.800	9.100	10.90	0	12.I	21.2
F FRANCE	1.055	8.247	57.59	36.40	65.50	107.8	2.9	12.6	53.4
D F.R.GERMANY	0.328	8.536	22.92	50.80	87.30	109.0	0.7	9.8	21.0
GB GREAT BRITAIN	4.070	7.952	14.12	59.80	76.10	74.50	6.8	10.5	19.0
NL HOLLAND	0.055	0.515	0.515	9.100	15.40	14.90	0.6	3.3	3.5
H HUNGARY	0	0	1.648	3.700	6.600	8.900	0	0	18.5
I ITALY	0.411	0.411	1.286	30.20	46.00	65.00	1.4	0.9	2.0
J JAPAN	0.150	14.36	28.10	59.80	137.9	209.0	0.3	10.4	13.4
ROK KOREA	0	0.557	7.203	2.500	9.400	21.20	0	5.9	34.0
RO ROMANIA	0	0	0.415	6.540	16.10	24.46	0	0	1.7
ZA SOUTH AFRICA	0	0	1.842	7.600	18.40	34.40	0	0	5.4
E SPAIN	0.153	1.073	7.531	17.92	29.90	42.88	0.9	3.6	17.6
S SWEDEN	0	3.700	9.440	15.40	26.50	32.80	0	14.0	28.8
CH SWITZERLAND	0.350	1.932	2.877	10.50	14.00	15.20	3.3	13.8	18.9
RW TAIWAN	0	1.208	4.884	2.720	9.056	17.68	0	13.3	27.6
USA USA	2.230	32.35	94.73	349.0	631.0	703.0	0.6	5.1	13.5
SU USSR	1.044	9.131	47.24	153.8	256.0	440.0	0.7	3.6	10.7
TOTALS	10.05	98.34	325.7	884.4	1587	2120	1.1	6.2	15.4

SOURCES — SEE REFERENCE [9] + UPDATE AT MID 1985

TABLE I GROWTH OF NUCLEAR CAPACITY

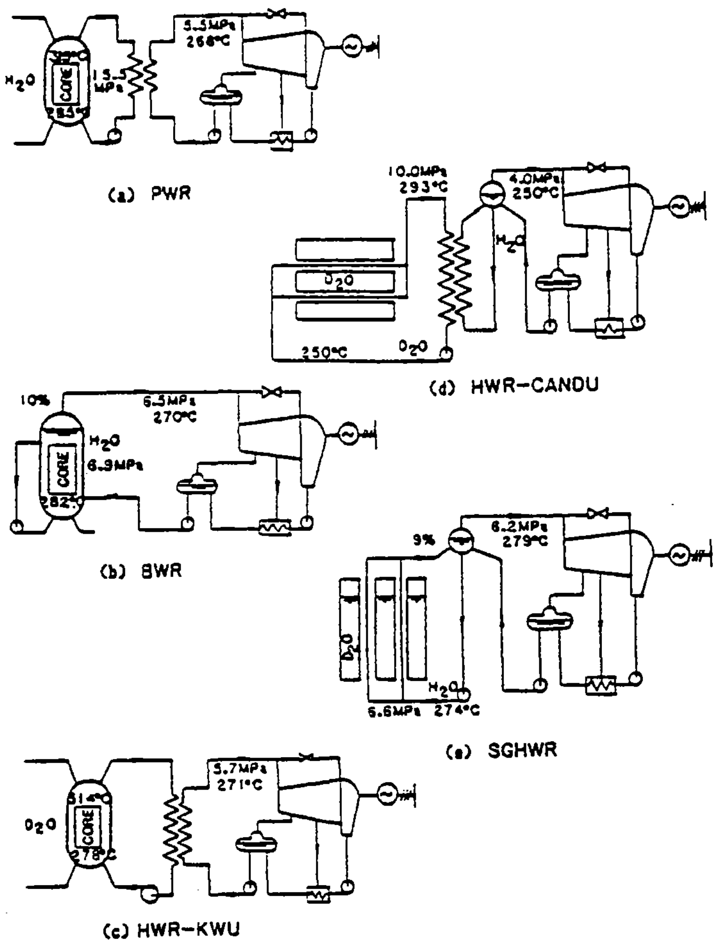


FIGURE 3 WATER REACTOR SYSTEMS

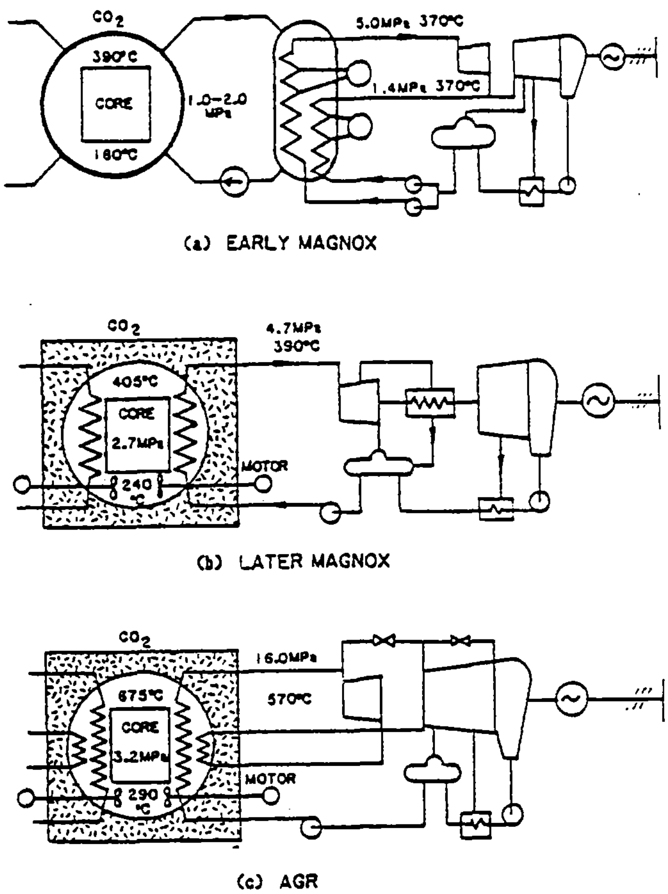


FIGURE 4 UK GAS COOLED REACTORS

Reactor Types	WATER REACTORS					GAS-COOLED REACTORS			OTHER TYPES
Form of Construction	Pressure Vessel			Pressure Tube		Pressure Vessel		High Temperature Reactor	
						Steel or Concrete	Concrete		
Concept	PWR (a)	BWR (b)	HWR-KWU (c)	HWR-CANDU (d)	SGHWR (e)	Magnox	AGR	HTGR or VHTR	
Moderator	H_2O	H_2O	D_2O	D_2O	D_2O	Graphite	Graphite	Graphite	
Coolant	H_2O	H_2O	D_2O	D_2O	H_2O	CO_2	CO_2	He	
Coolant Phase	Pressurized	Boiling	Pressurized	Pressurized	Boiling	Gas	Gas	Gas	
Cycle	Indirect	Direct	Indirect	Indirect	Direct	Indirect	Indirect	Indirect	
Fuel and Enrichment %	UO2 2.0-4.0	UO2 2.1-2.8	UO2 Nat U	UO2 Nat U	UO2 2.3	U-nat	UO2 2.1-2.7	U/Th ~ 93	
Worldwide Operational Capacity at end 1983 * GWe	108	46	0.4 (-)	8.2 (4.3)	0.1 (-)	7.5 (3.9)	2.6 (1.4)	0.4 (-)	15.2
(%)	(56.5)	(24.1)	10.7 (5.6) covers all HWR plant			11.1 (5.8) covers all GCR plant			(3.0)

* Numerical Data from Reference (2)

TABLE 2 PRINCIPAL REACTOR SYSTEMS

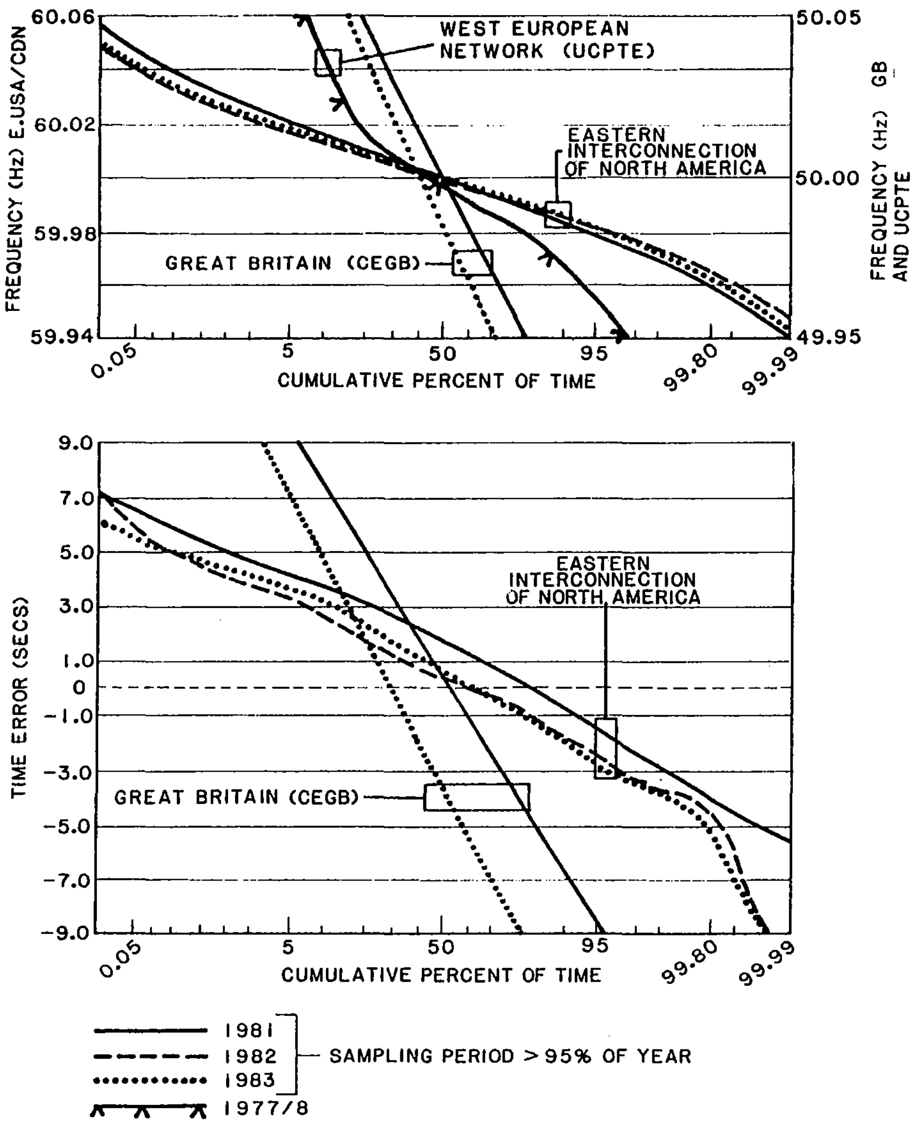


FIGURE 5 CUMULATIVE DISTRIBUTIONS OF FREQUENCY AND TIME ERROR

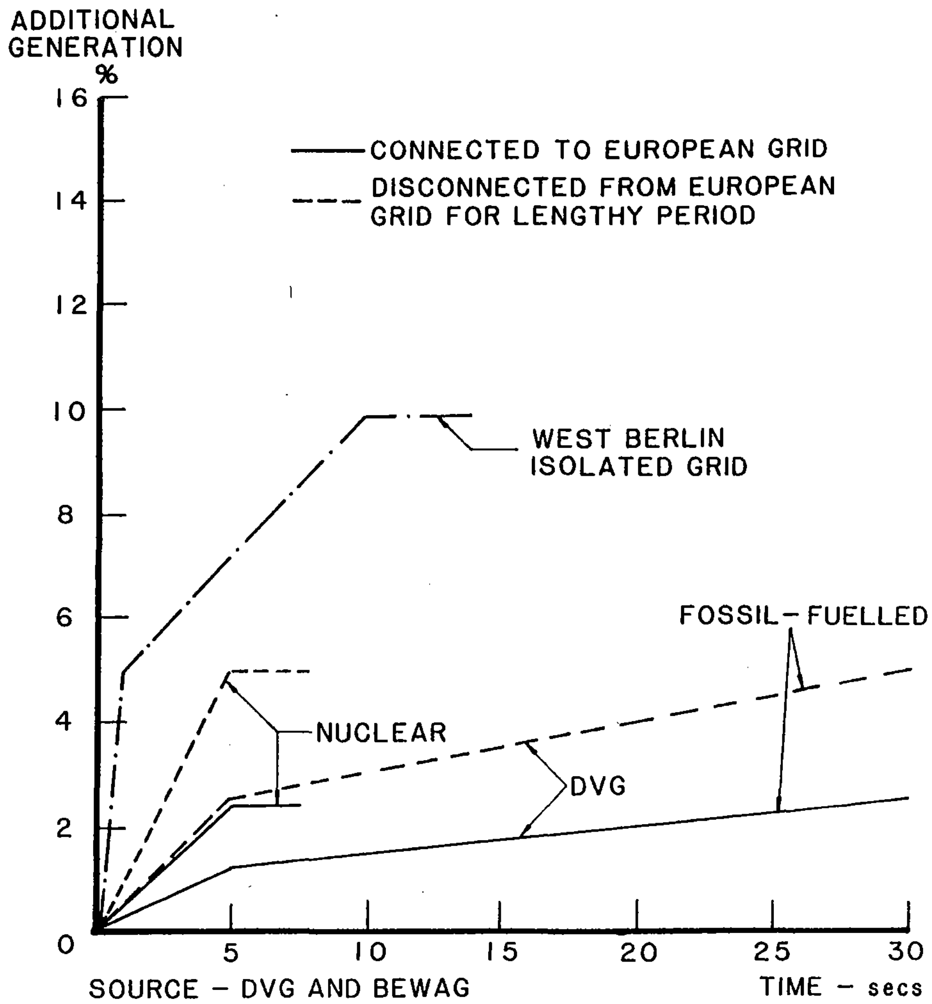


FIGURE 6 REQUIREMENTS FOR PRIMARY RESERVE IN THE FEDERAL REPUBLIC OF GERMANY

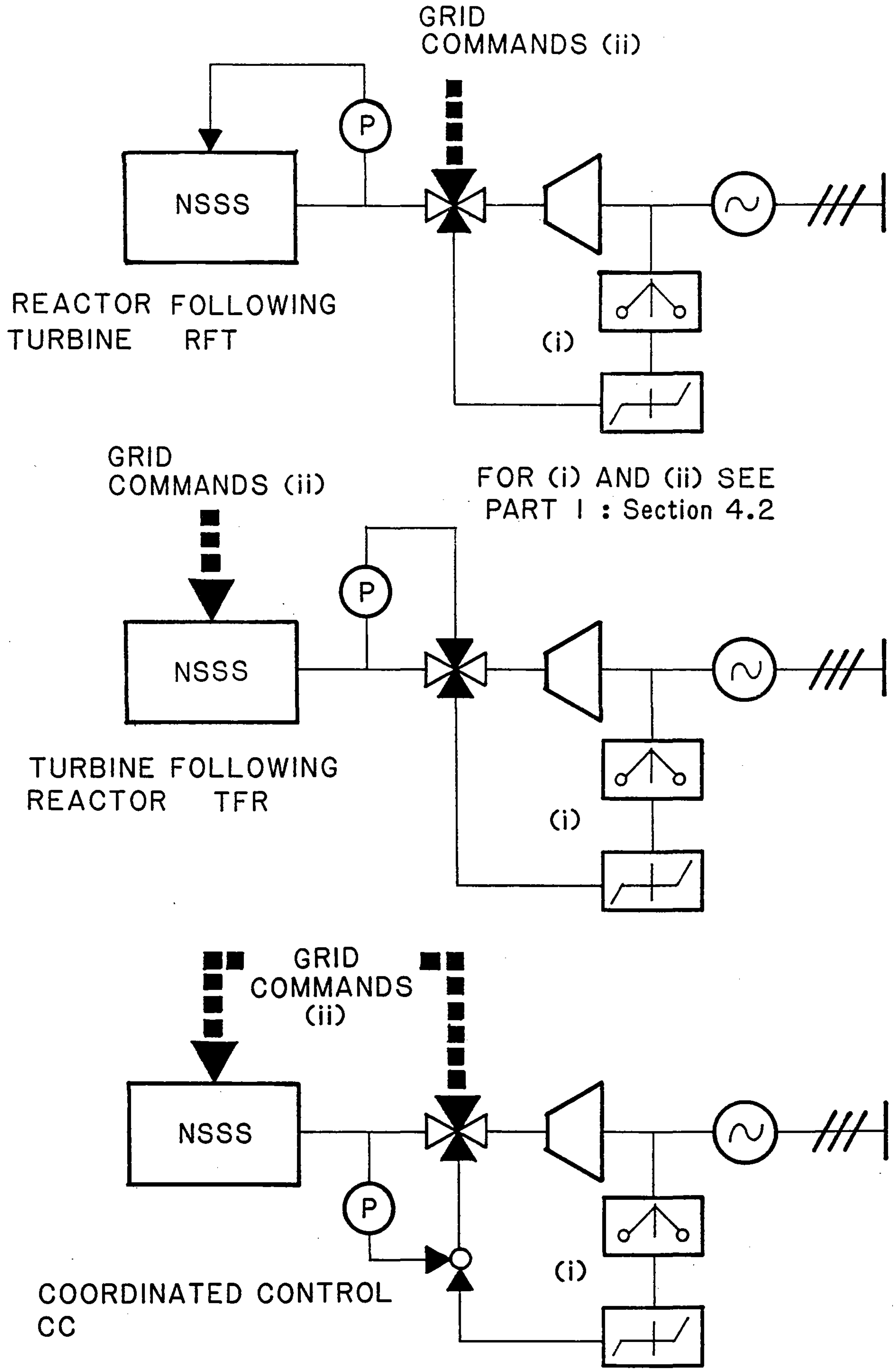
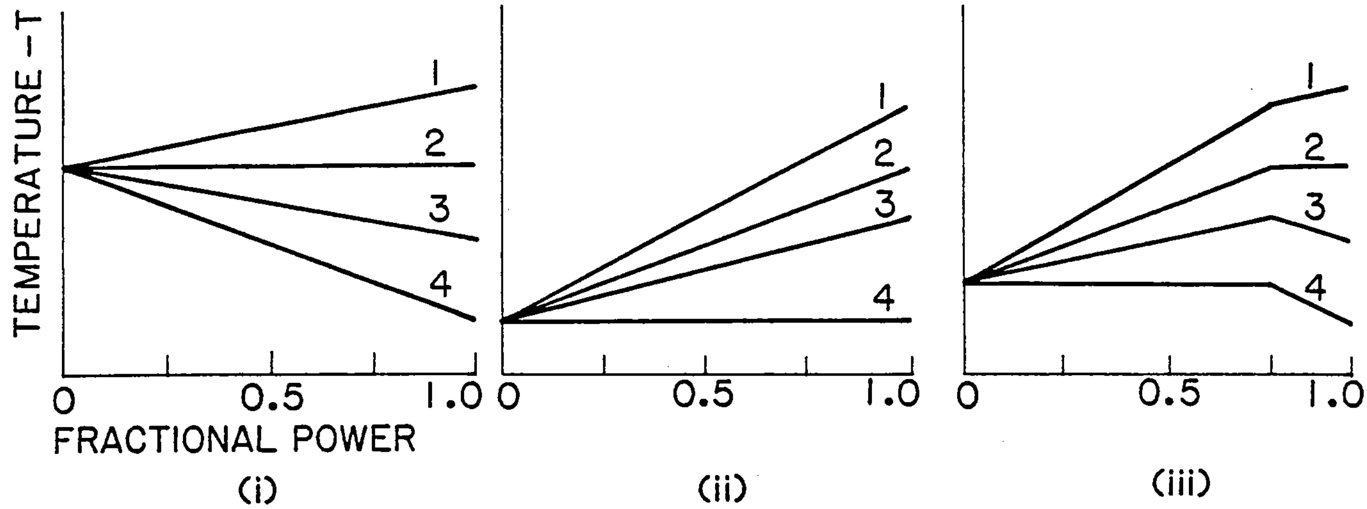
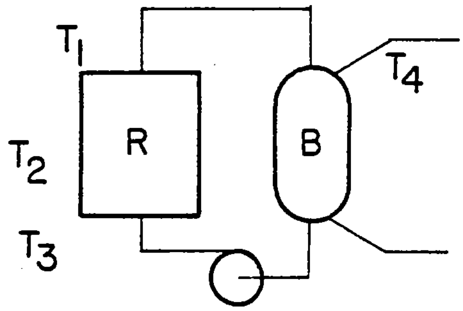


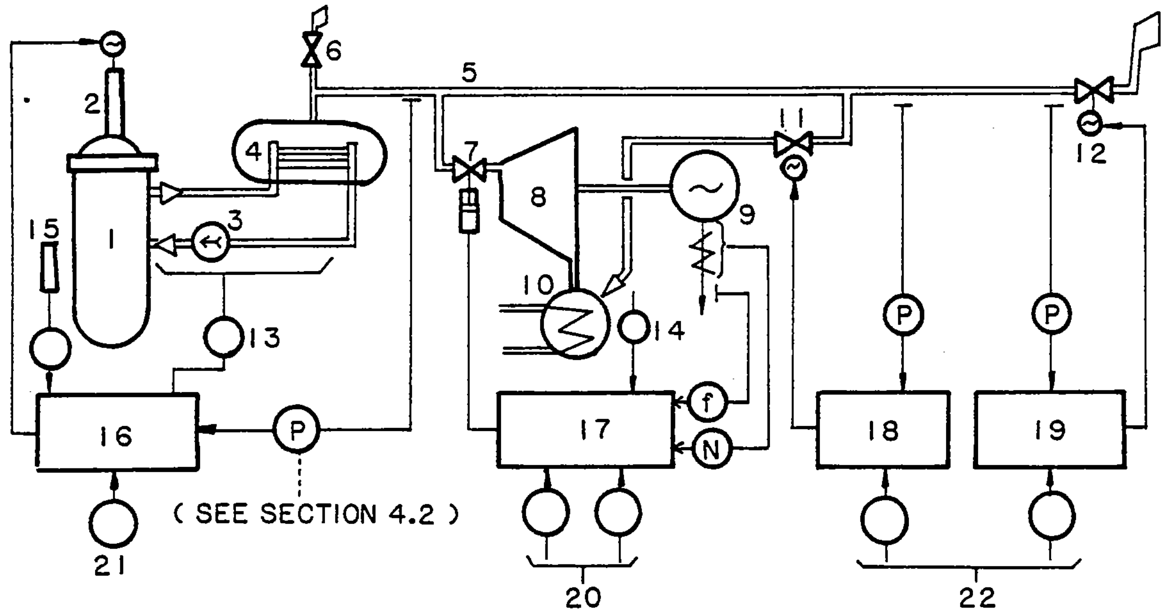
FIGURE 7 NUCLEAR PLANT CONTROL STRATEGIES





1. PRIMARY COOLANT TEMPERATURE AT REACTOR OUTLET T_1
2. AVERAGE COOLANT TEMPERATURE T_2
3. PRIMARY COOLANT TEMPERATURE AT REACTOR INLET T_3
4. FINAL STEAM TEMPERATURE IN STEAM GENERATOR T_4

(a) STEADY STATE SCHEDULES FOR RUSSIAN PWR's FROM REFERENCE [31]



1. REACTOR
2. REACTOR CONTROL MECHANISM
3. MAIN CIRCULATION PUMP
4. STEAM GENERATOR
5. MAIN STEAM HEADER
6. SAFETY VALVE
7. TURBINE GOVERNING VALVES
8. TURBINE
9. GENERATOR
10. CONDENSER
11. RRD–C
12. RRD–A
13. REACTOR POWER LIMITATION
14. TURBINE LOAD LIMITATION
15. IONISATION CHAMBERS
16. REACTOR POWER REGULATOR
17. TURBINE LOAD REGULATOR
18. RRD–C REGULATOR
19. RRD–A REGULATOR
20. FROM POWER SYSTEM OR STATION REGULATOR (RFT)
21. COMMAND (TFR)
22. PRESSURE SETPOINTS.

(b) STRUCTURAL DIAGRAM OF UNIT POWER REGULATION SYSTEM FOR LOAD-FOLLOWING DUTY (RFT MODE).

FIGURE 8 CONTROL SYSTEMS FOR RUSSIAN VVER REACTORS

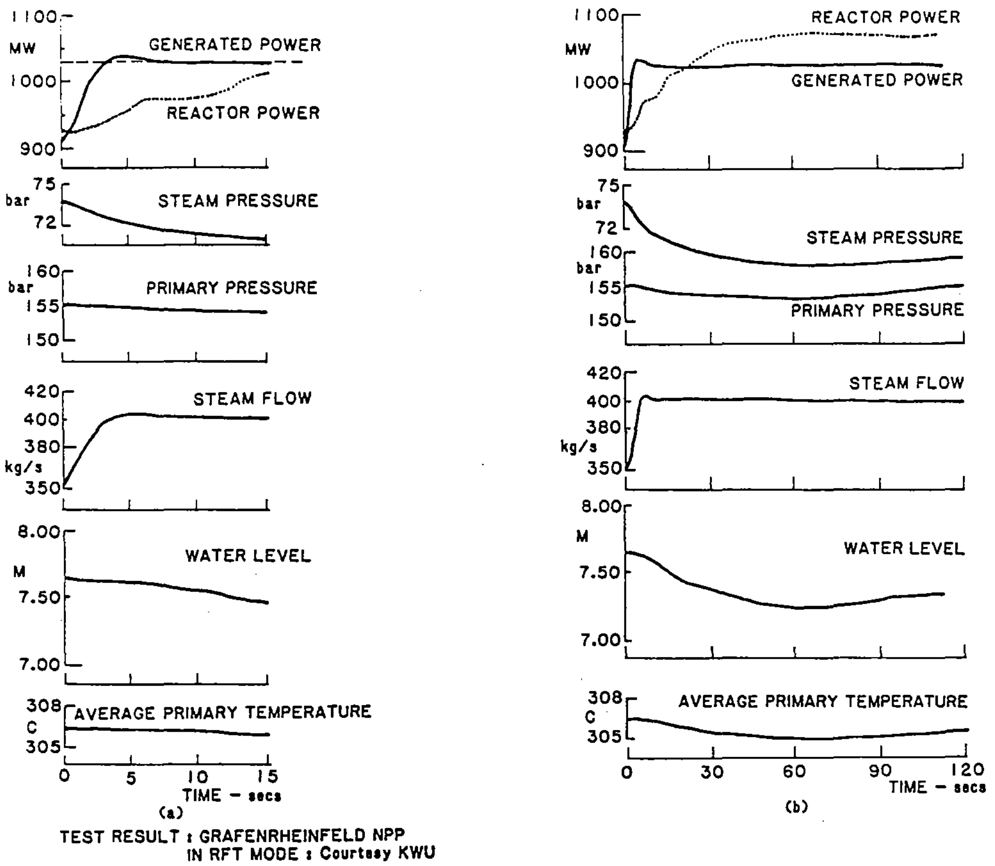


FIGURE 9 SHORT TERM AND LONG TERM RESPONSE OF A 1300MW PWR TO A +10% STEP IN POWER DEMAND

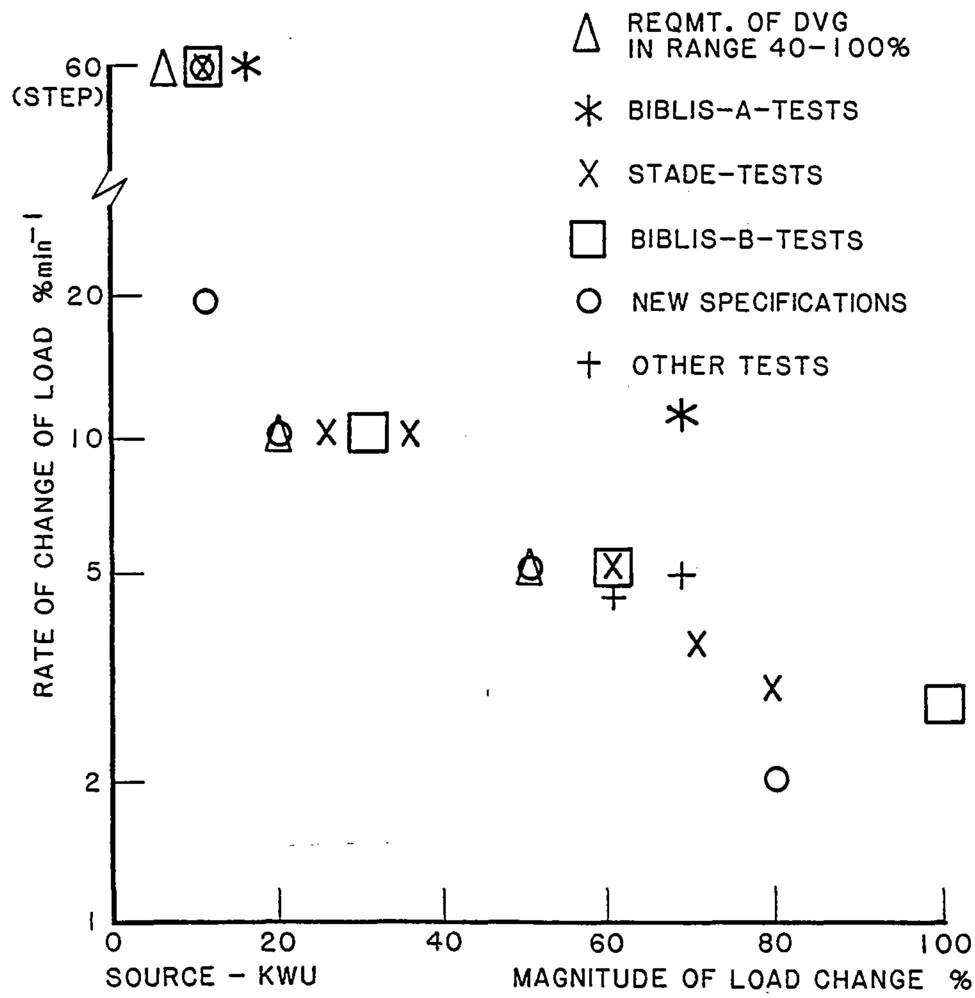


FIGURE 10 LOADING RATES FOR PWR PLANTS

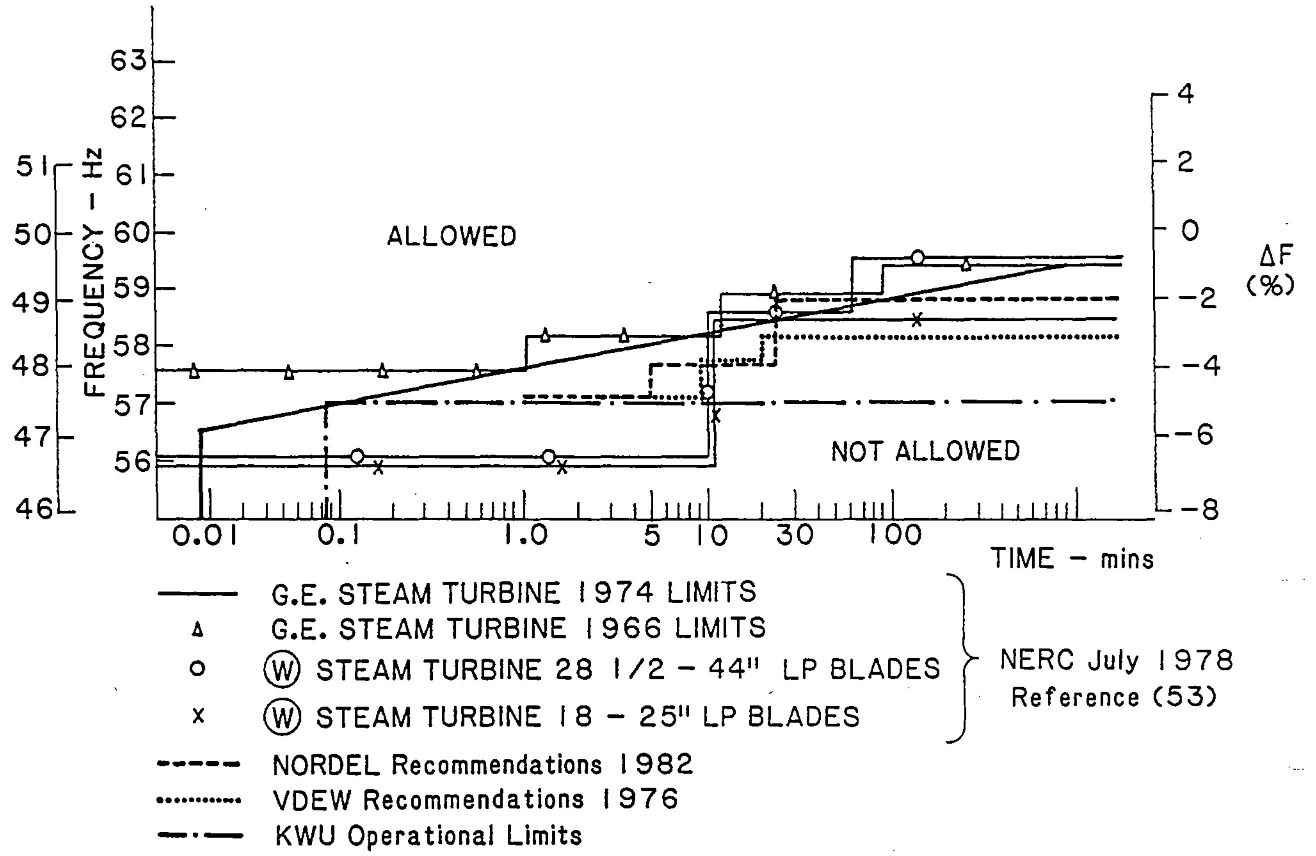


FIGURE 11 UNDERFREQUENCY vs TIME-TO-DAMAGE FOR TURBINE BLADING

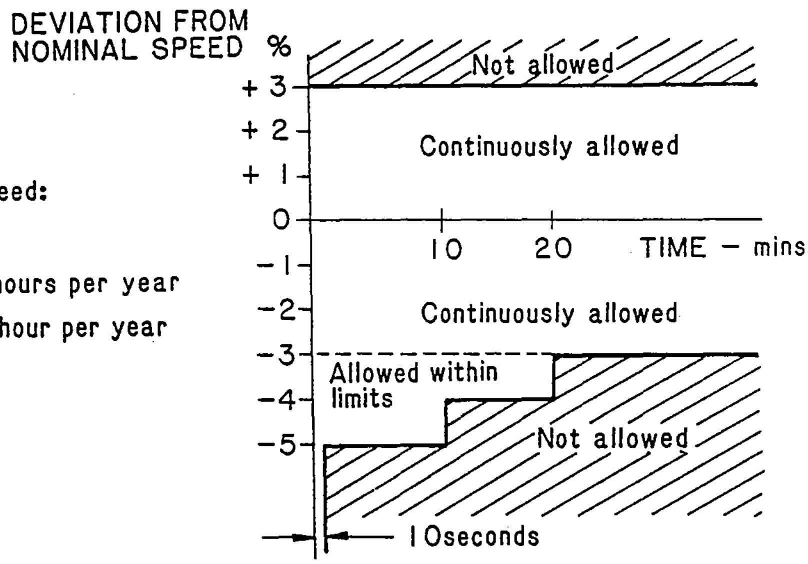
Allowable deviations from the nominal speed:

from +3 to −3% Continuously allowed

to −4% 20 minutes, total 2 hours per year

to −5% 10 minutes, total 1 hour per year

exceeding −5% 10 seconds



(BROWN BOVERI CORPORATION)

FIGURE 12 OPERATION OF STEAM TURBINES AT SPEEDS DEVIATING FROM THE NOMINAL SPEED

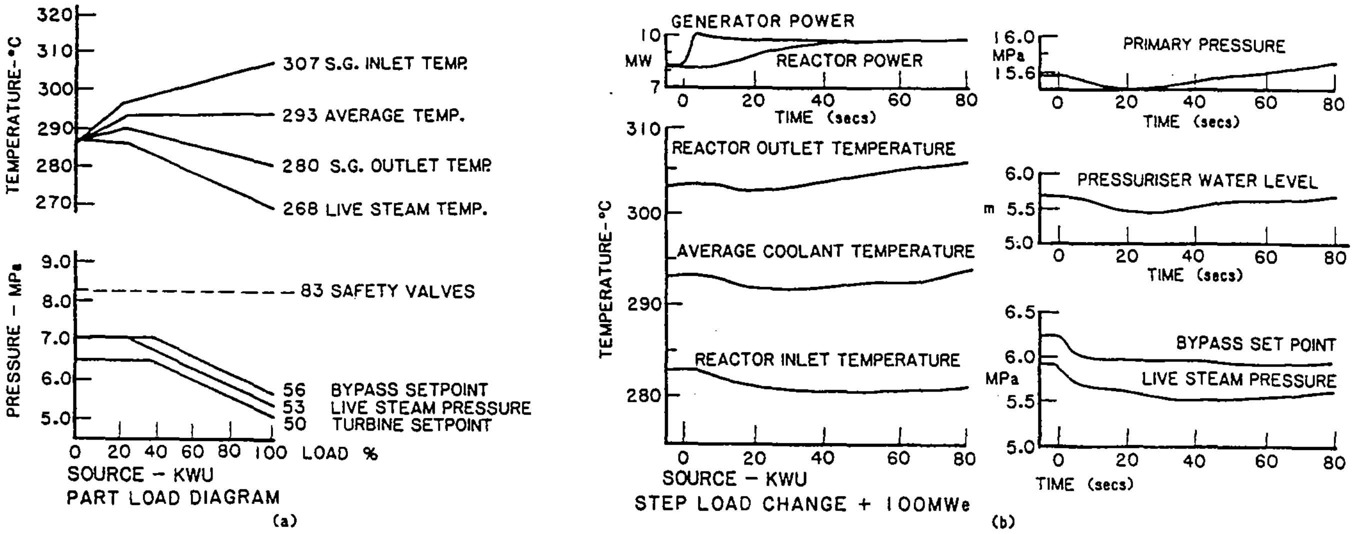


FIGURE 13 PERFORMANCE OF A TYPICAL PWR - 660MWe

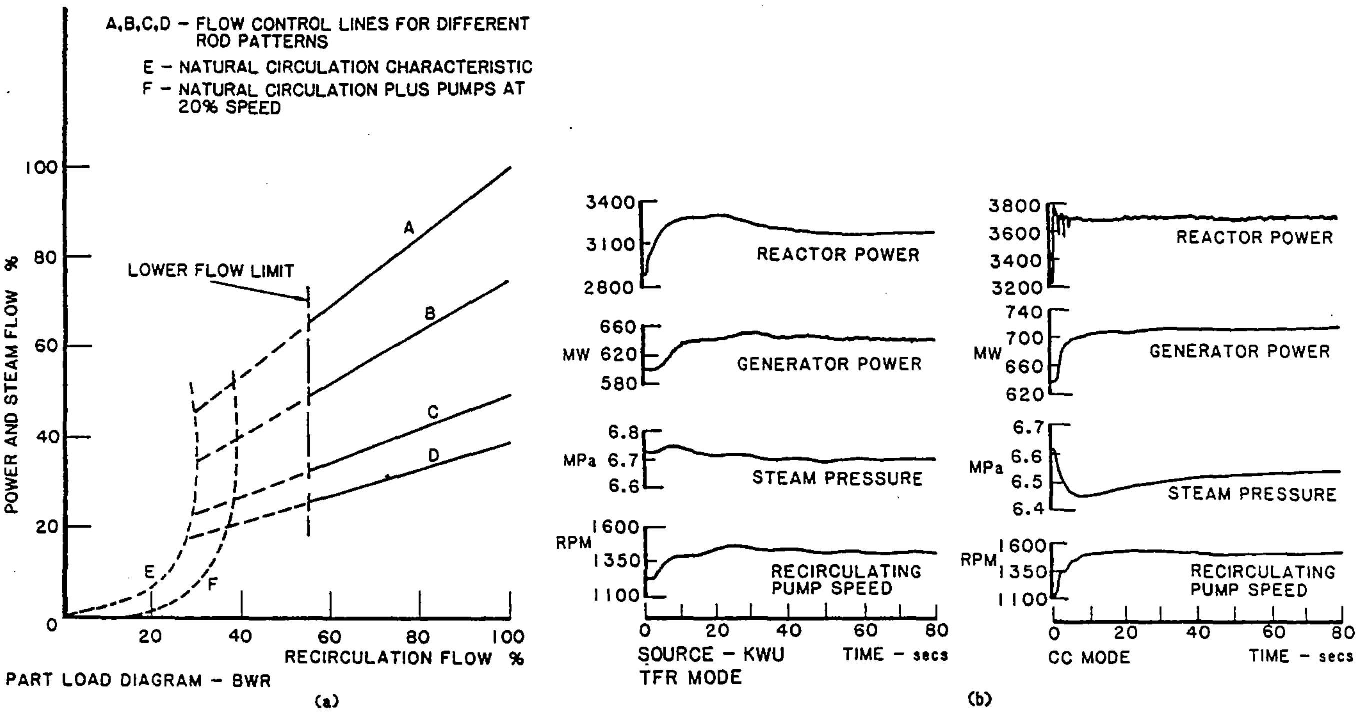


FIGURE 14 PERFORMANCE OF A TYPICAL BWR - 660MWe

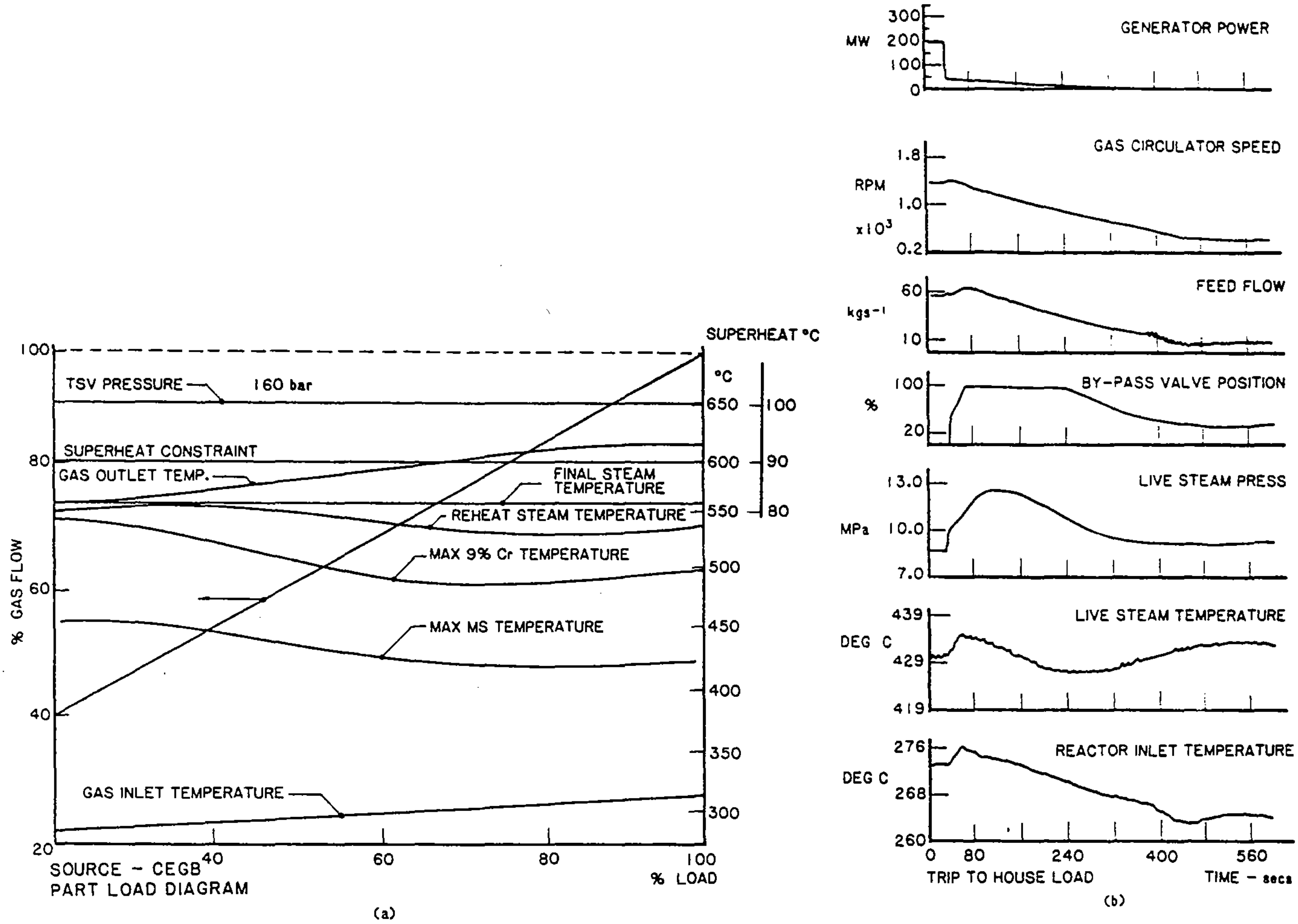


FIGURE 15 PERFORMANCE OF A TYPICAL AGR - 660MWe

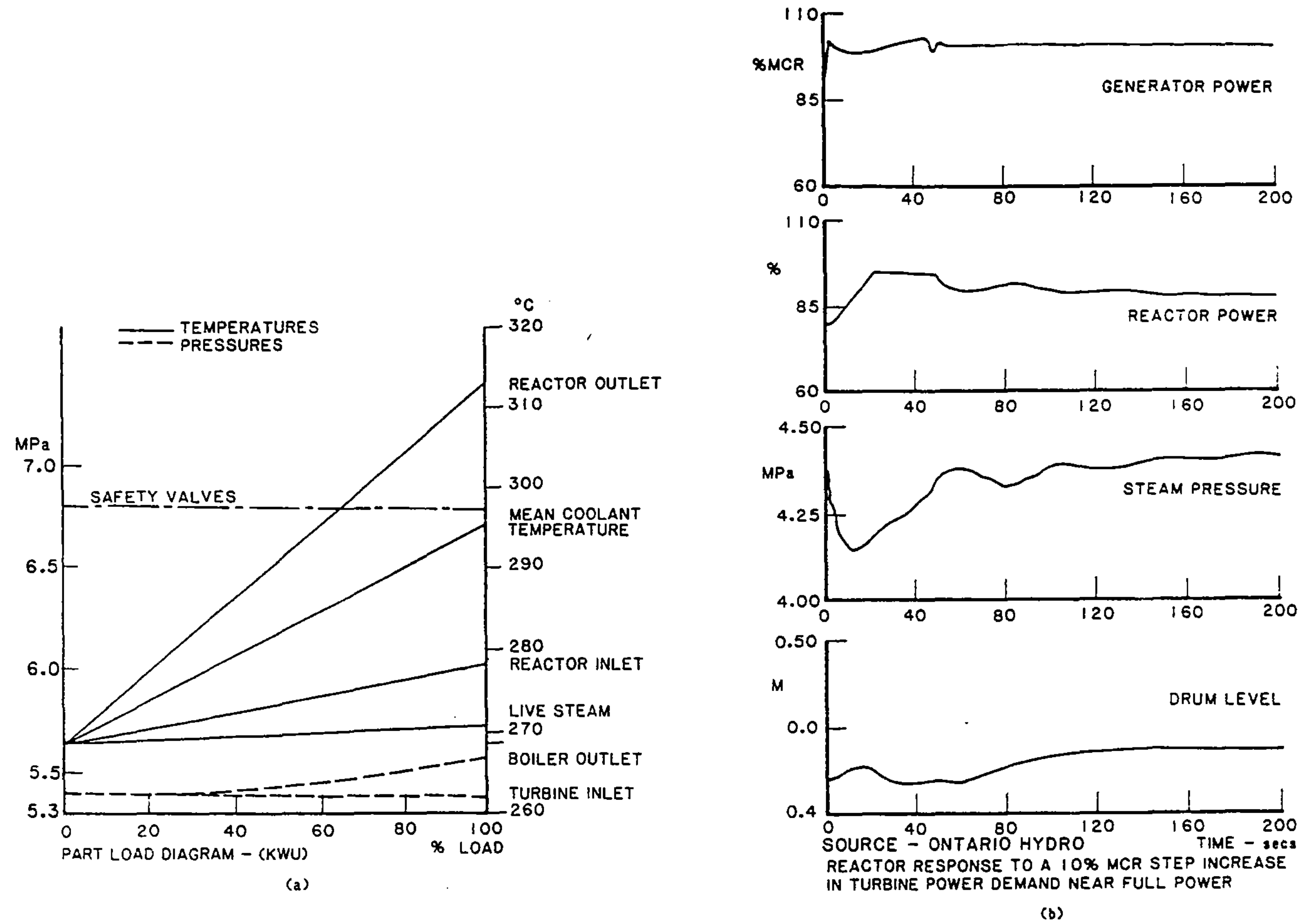


FIGURE 16 PERFORMANCE OF A TYPICAL HWR - 850MWe

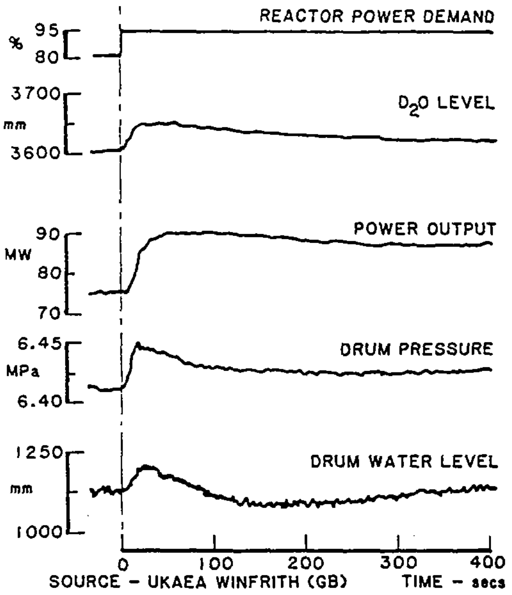


FIGURE 17 RESPONSE OF A 100MW SGHWR IN TFR MODE

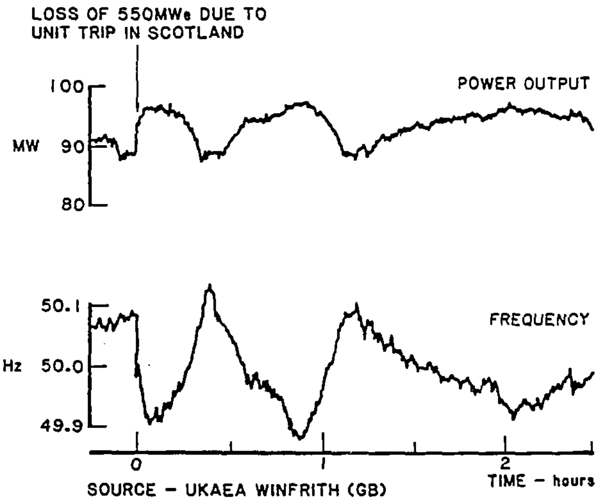


FIGURE 18 LOAD FOLLOWING IN RFT MODE OF A 100MW SGHWR IN SOUTHERN ENGLAND

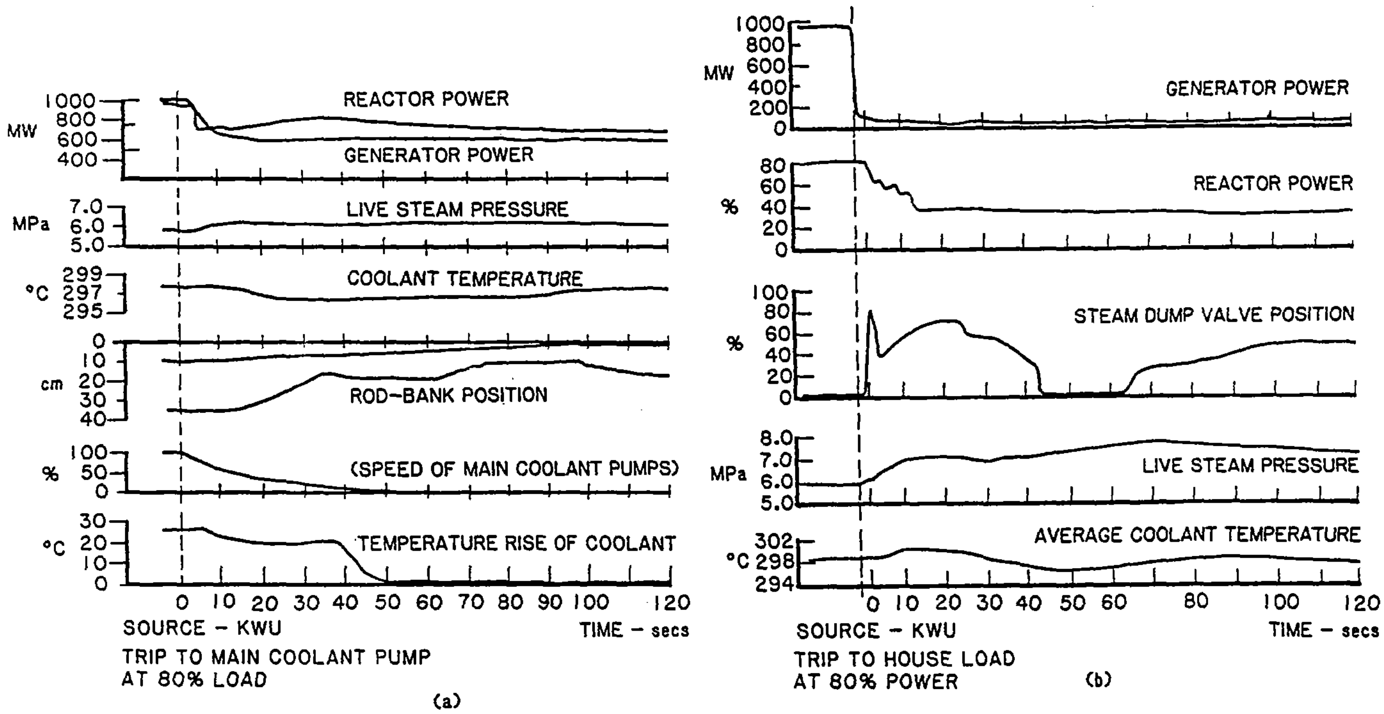


FIGURE 19 RESPONSE OF A TYPICAL PWR TO MAJOR DISTURBANCES – 1200MWe

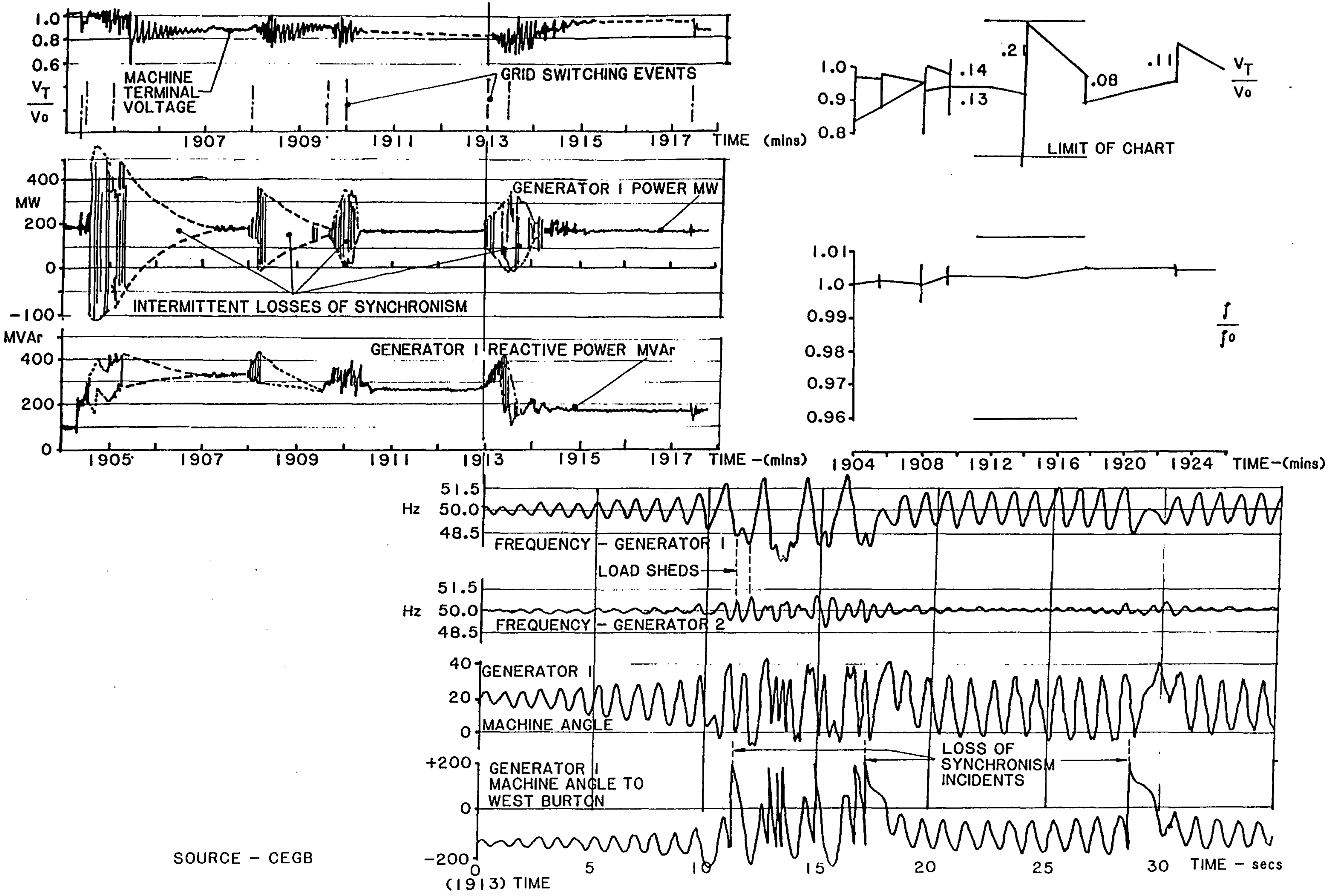
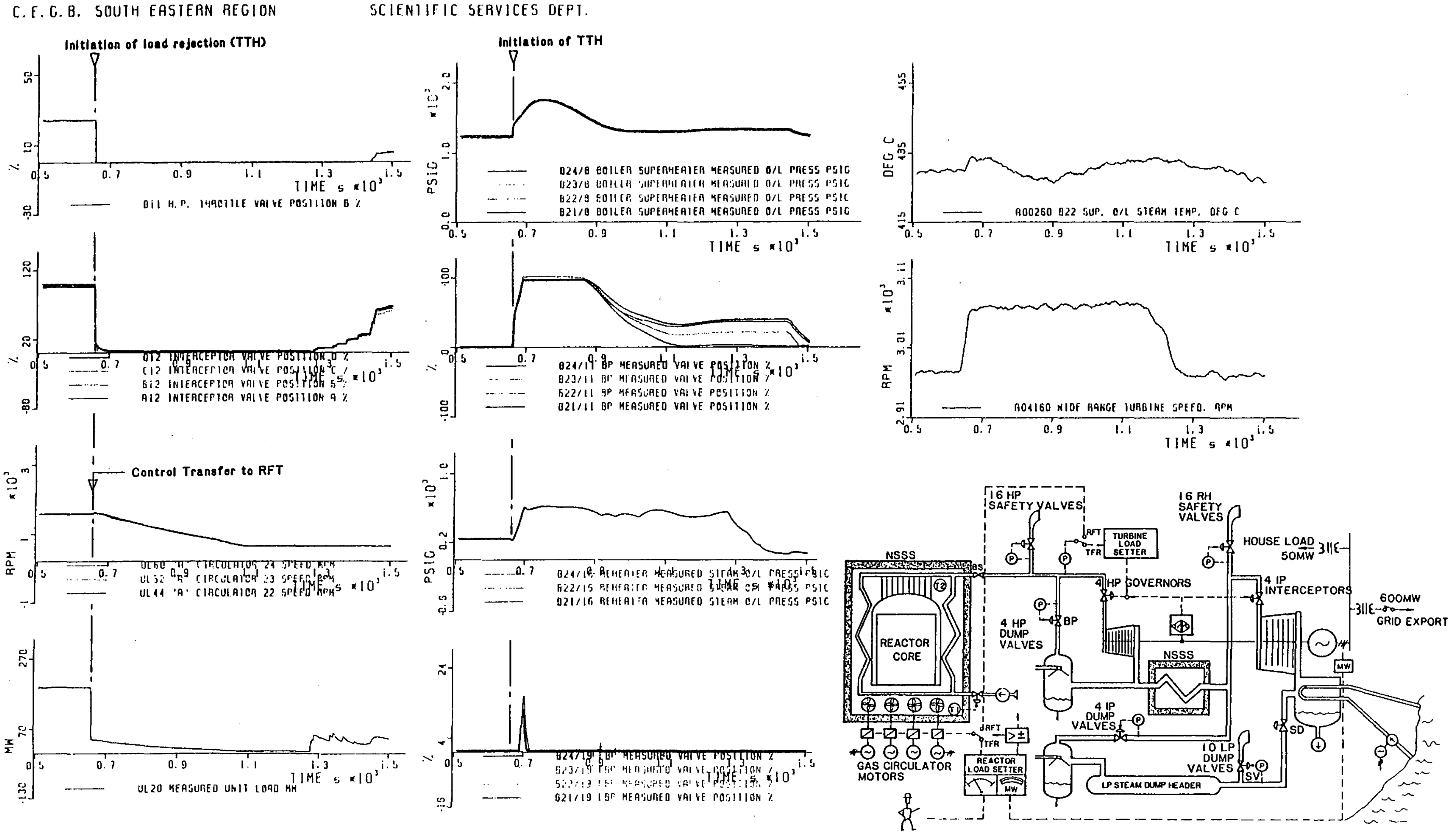


FIGURE 25 GRID EVENTS ON A MAGNOX STATION IN S E ENGLAND: 28 NOVEMBER 1980

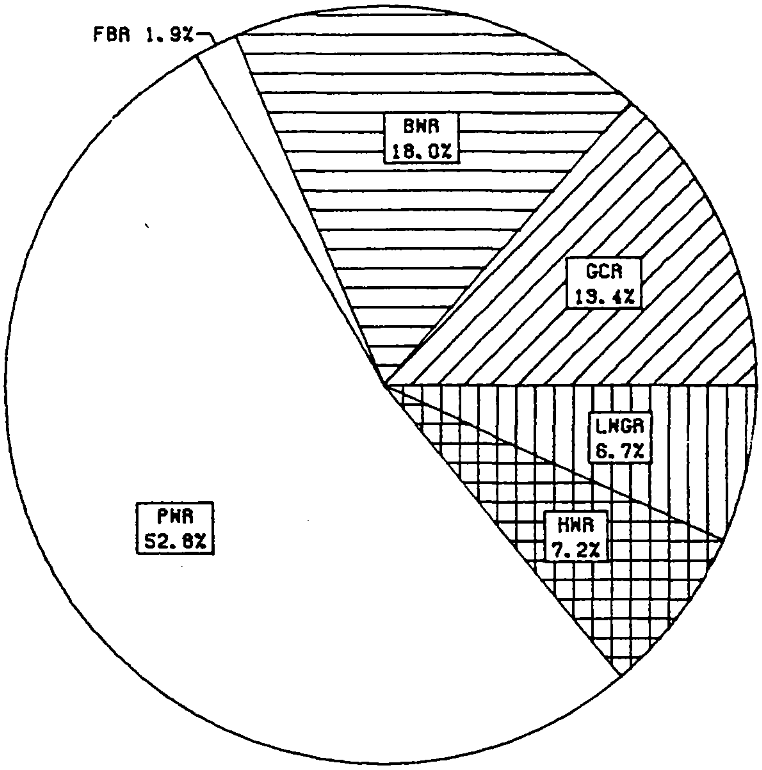


Courtesy - CEGB

FIGURE 26 LOAD REJECTION TEST: RESULTS FROM DUNGENESS 'B' HIGH SPEED DATA LOGGER ON 18.8.83

NUMBER OF REACTORS BY TYPE
NUMBER OF REACTORS IN SURVEY : 373

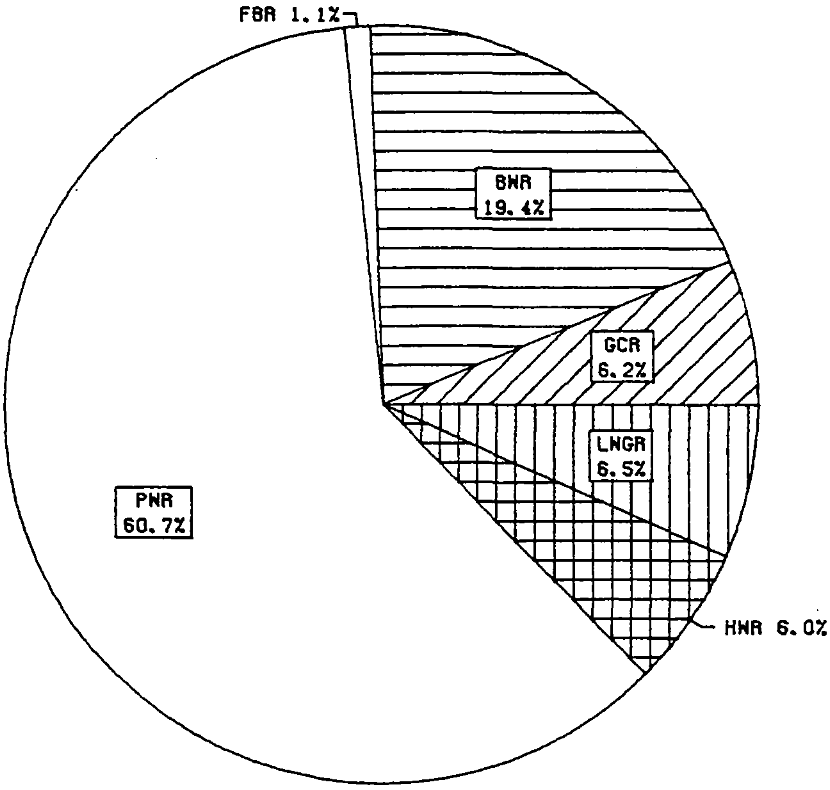
FIG. 27



SOURCE CIGRE SC 39-04 : 2/5/85

MW OUTPUT BY REACTOR TYPE IN SURVEY
TO 1990 TOTAL OUTPUT : 296782 MW

FIG .28



SOURCE CIGRE SC 39-04 : 2/5/85

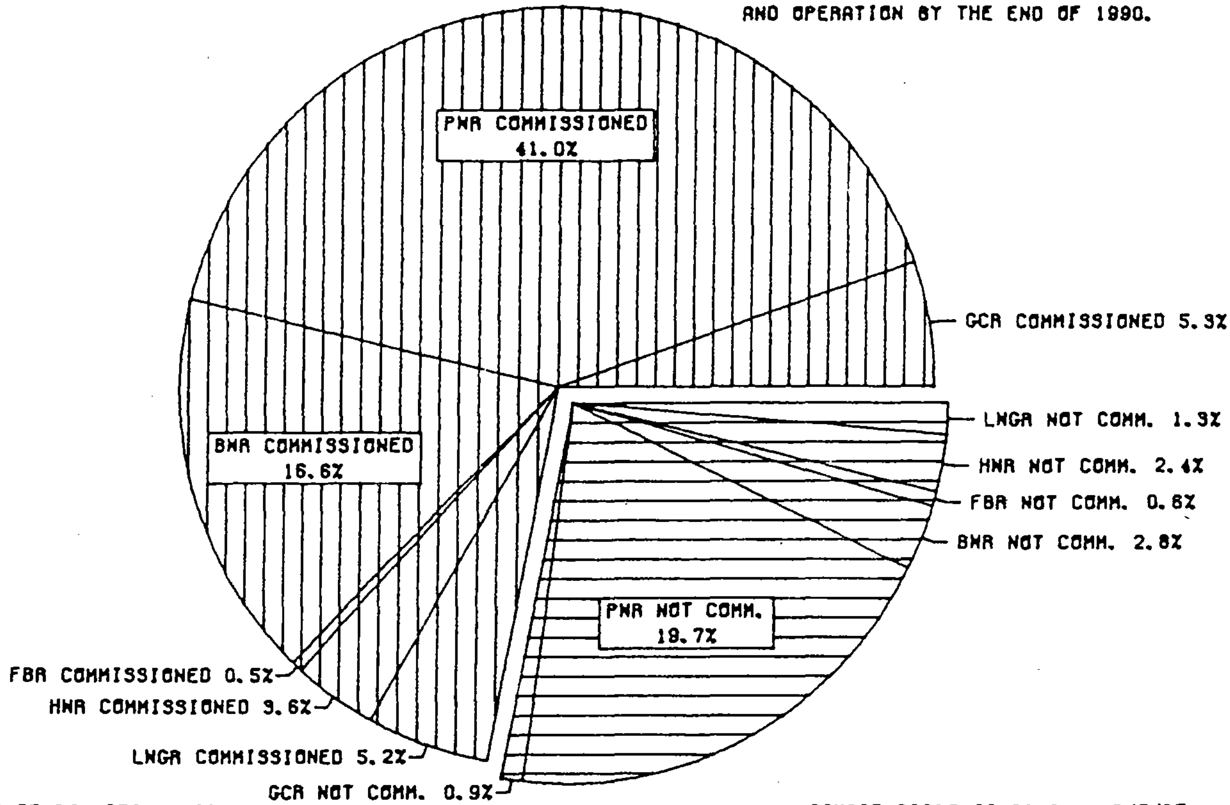
POWER GENERATION IN THIS SURVEY AT 1985

NSSS IN OPERATION & IN CONSTRUCTION

TO 1990 TOTAL OUTPUT : 296782 MW

FIG. 29

NOTE : PLANT IN CONSTRUCTION CONFINED TO NSSS WHICH ARE TARGETED FOR COMMISSIONING AND OPERATION BY THE END OF 1990.



SAMPLE OF REPLIES = 100%

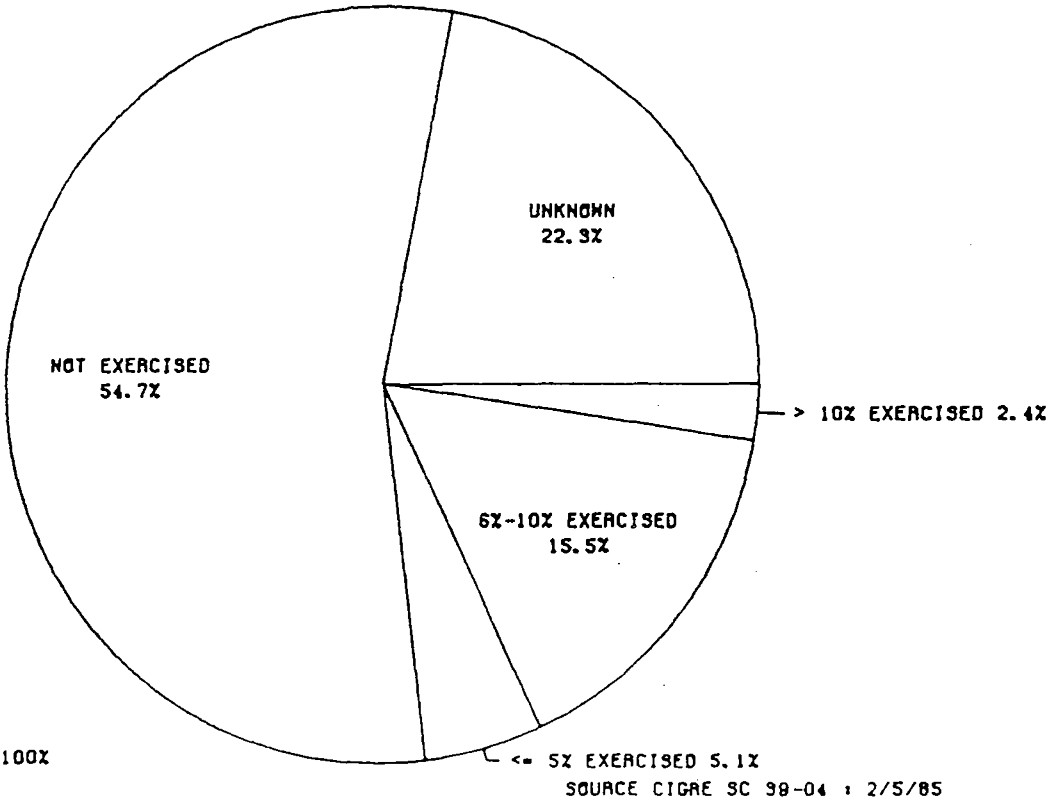
SOURCE CIGRE SC 39-04 : 2/5/85

REGULATING BAND EXERCISED BY

PRIMARY CONTROL

NUMBER OF REACTORS IN SURVEY : 373

FIG. 30

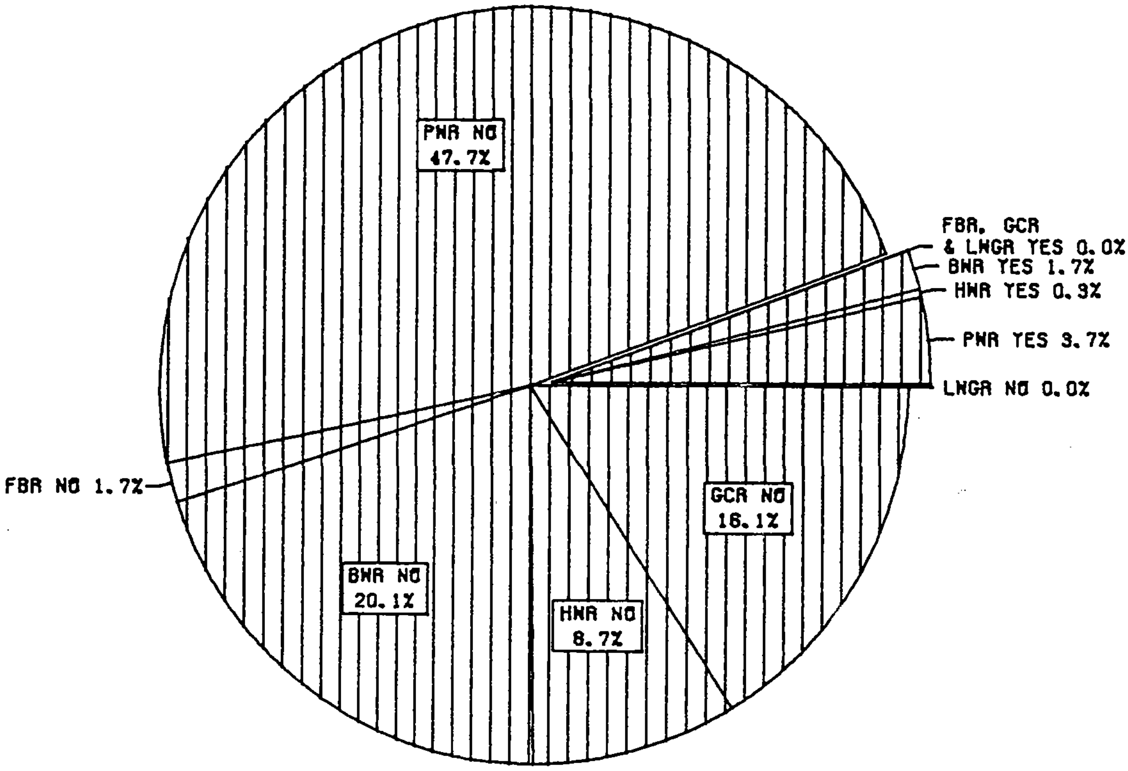


SAMPLE OF REPLIES = 100%

SOURCE CIGRE SC 39-04 : 2/5/85

SECONDARY CONTROL
UTILISATION - NOW
NUMBER OF REACTORS IN SURVEY : 299

FIG. 35

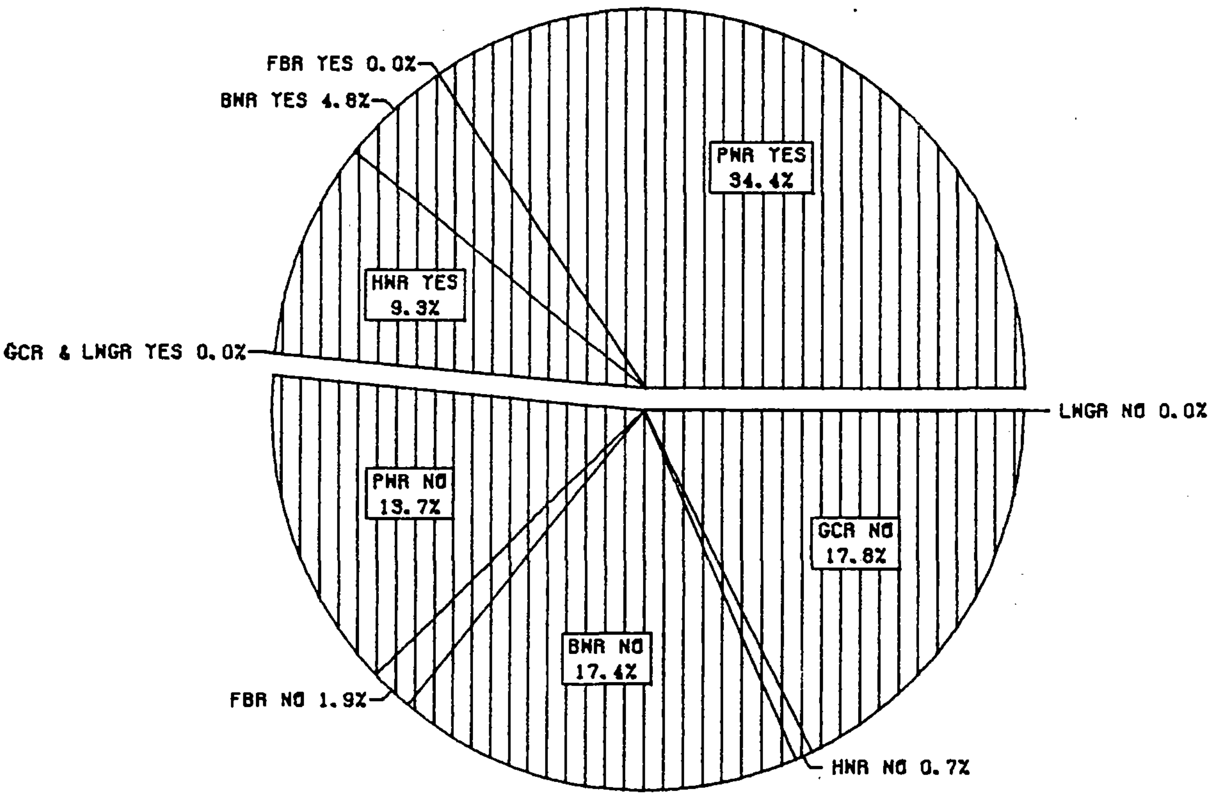


SAMPLE OF REPLIES = 80.2%

SOURCE CIGRE SC 39-04 : 2/5/85

SECONDARY CONTROL
UTILISATION - FUTURE
NUMBER OF REACTORS IN SURVEY : 271

FIG. 36



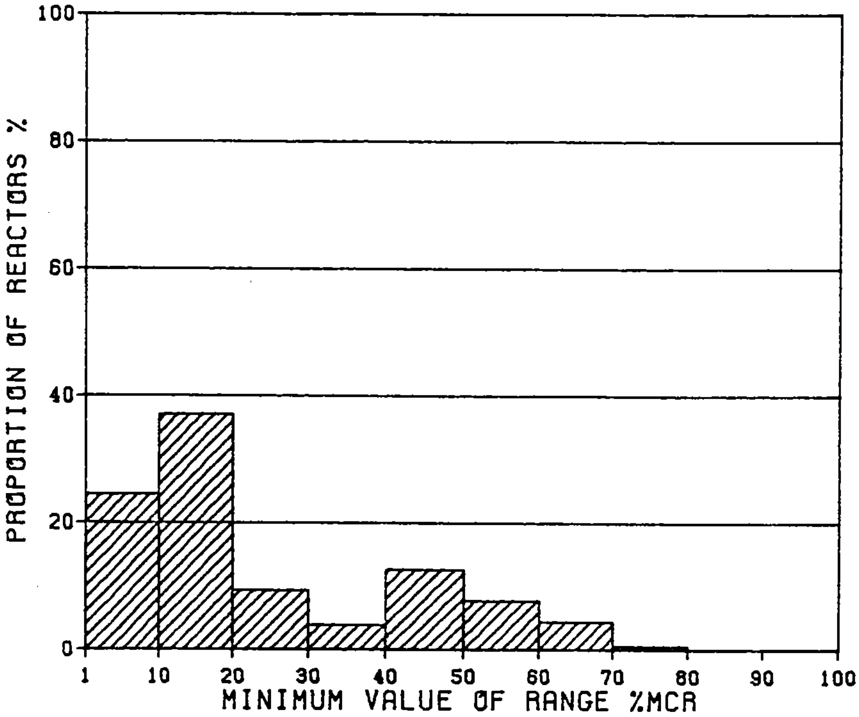
SAMPLE OF REPLIES = 72.7%

SOURCE CIGRE SC 39-04 : 2/5/85

AUTO CONTROL RANGE %MCR

NUMBER OF REACTORS IN SURVEY : 364

FIG. 37



NOTE : MAXIMUM OF RANGE = 100%

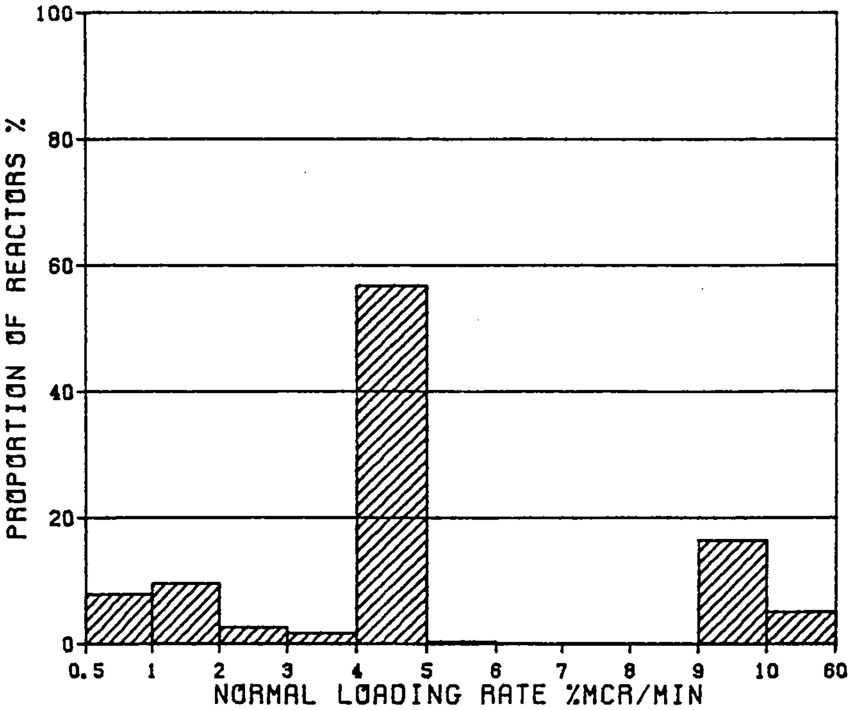
SAMPLE OF REPLIES = 97.6%

SOURCE CIGRE SC 39-04 : 2/5/85

NORMAL LOADING RATE %MCR PER MIN

NUMBER OF REACTORS IN SURVEY : 354

FIG. 38



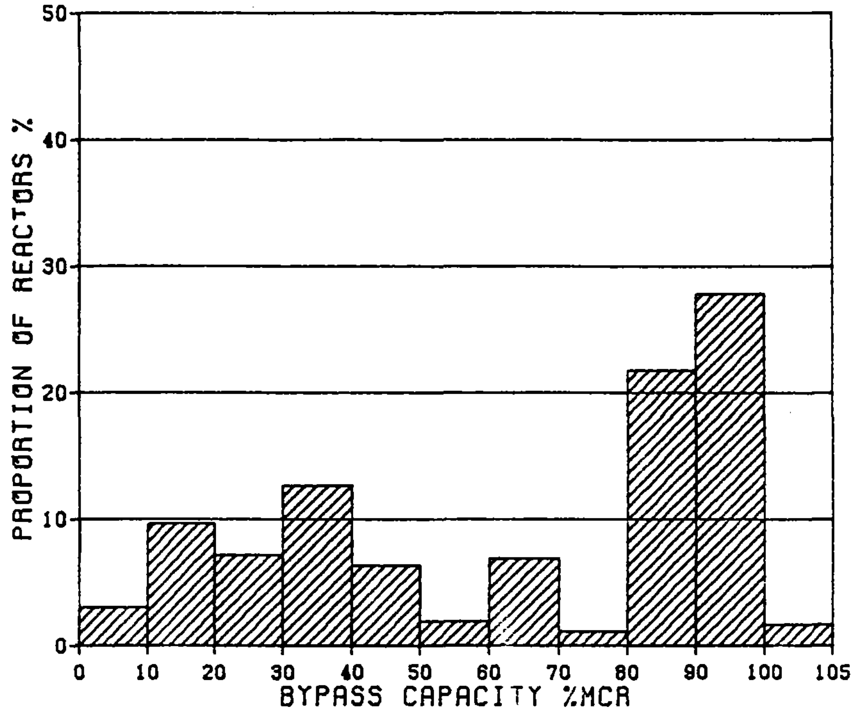
NOTE: LOADING RATE IS INFLUENCED BY
AMPLITUDE OF LOAD CHANGE. FOR
SMALL CHANGES FASTER RATES APPLY.
REFER FIGURE 10

SAMPLE OF REPLIES = 94.9%

SOURCE CIGRE SC 39-04 : 2/5/85

PROVISION OF TURBINE BYPASS CAPACITY
(TO CONDENSER) NUMBER OF REACTORS IN SURVEY : 363

FIG. 39

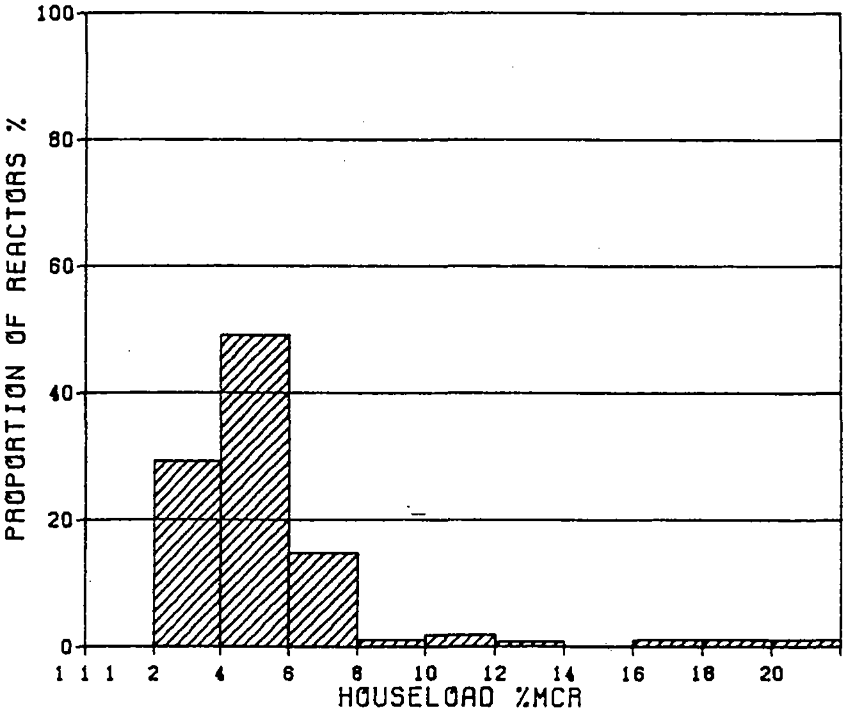


SAMPLE OF REPLIES = 97.3%

SOURCE CIGRE SC 39-04 : 2/5/85

HOUSELOAD - %MCR
NUMBER OF REACTORS IN SURVEY : 373

FIG. 40



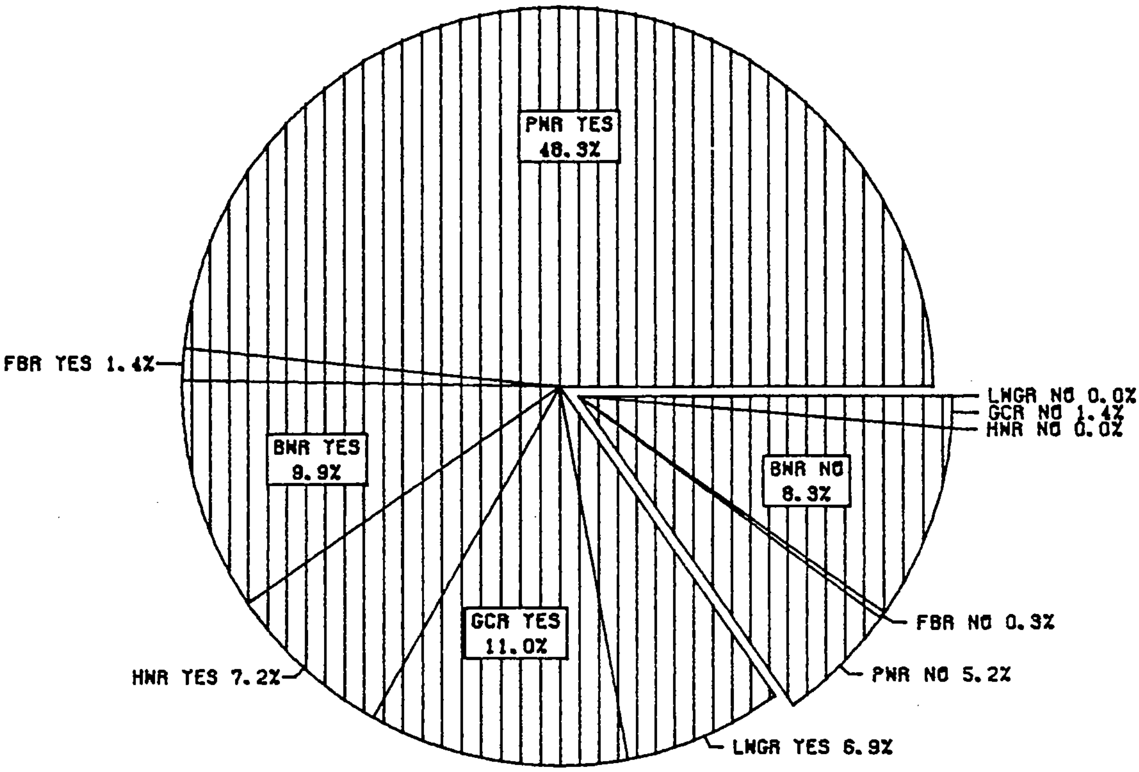
SAMPLE OF REPLIES = 100%

SOURCE CIGRE SC 39-04 : 2/5/85

ISOLATED OPERATION CAPABILITY

NUMBER OF REACTORS IN SURVEY : 363

FIG. 41



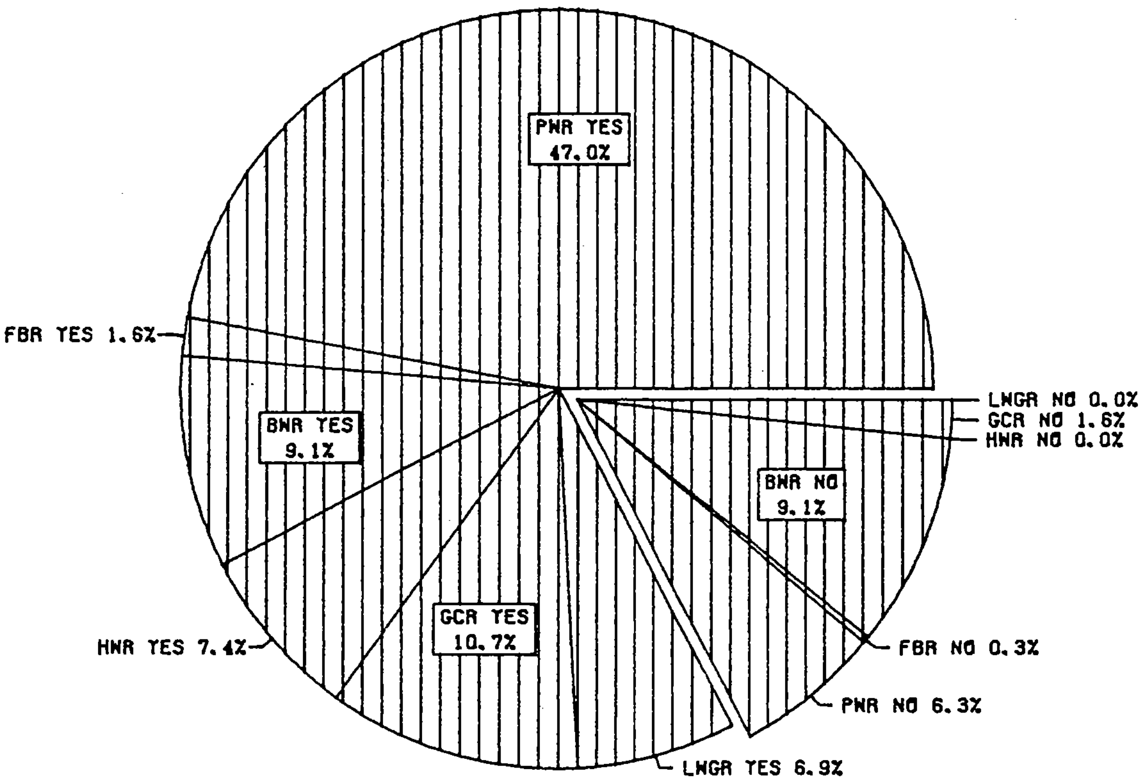
SAMPLE OF REPLIES = 97.3%

SOURCE CIGRE SC 39-04 : 2/5/85

CAPABILITY OF TTH

NUMBER OF REACTORS IN SURVEY : 364

FIG. 42



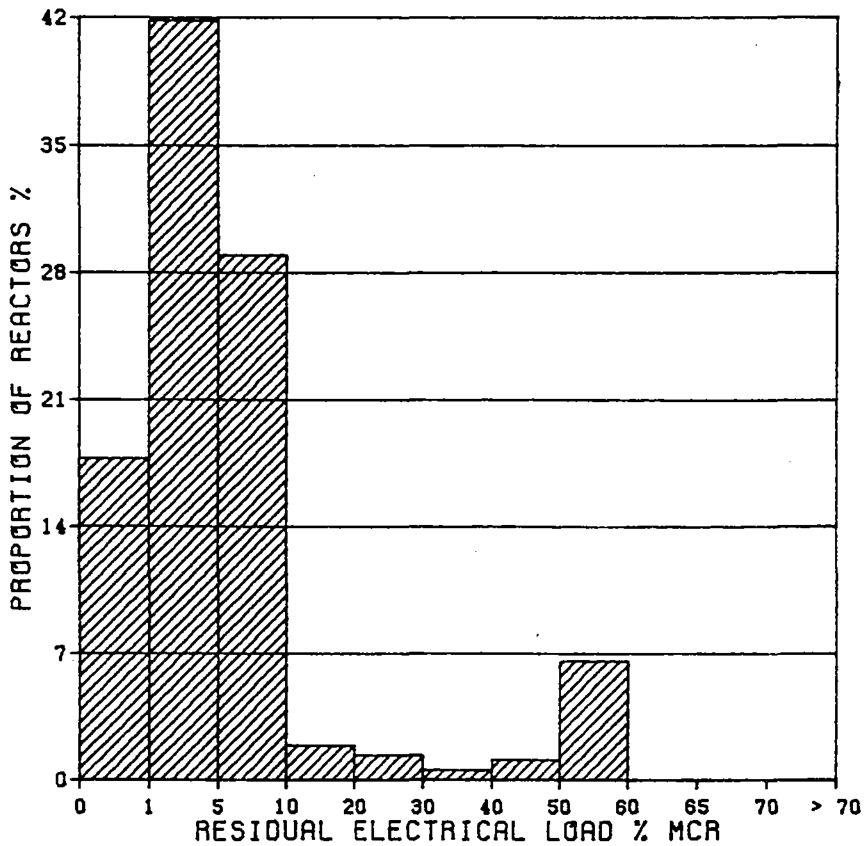
SAMPLE OF REPLIES = 97.6 %

SOURCE CIGRE SC 39-04 : 2/5/85

CAPABILITY OF TTH

NUMBER OF REACTORS IN SURVEY : 365

FIG. 43



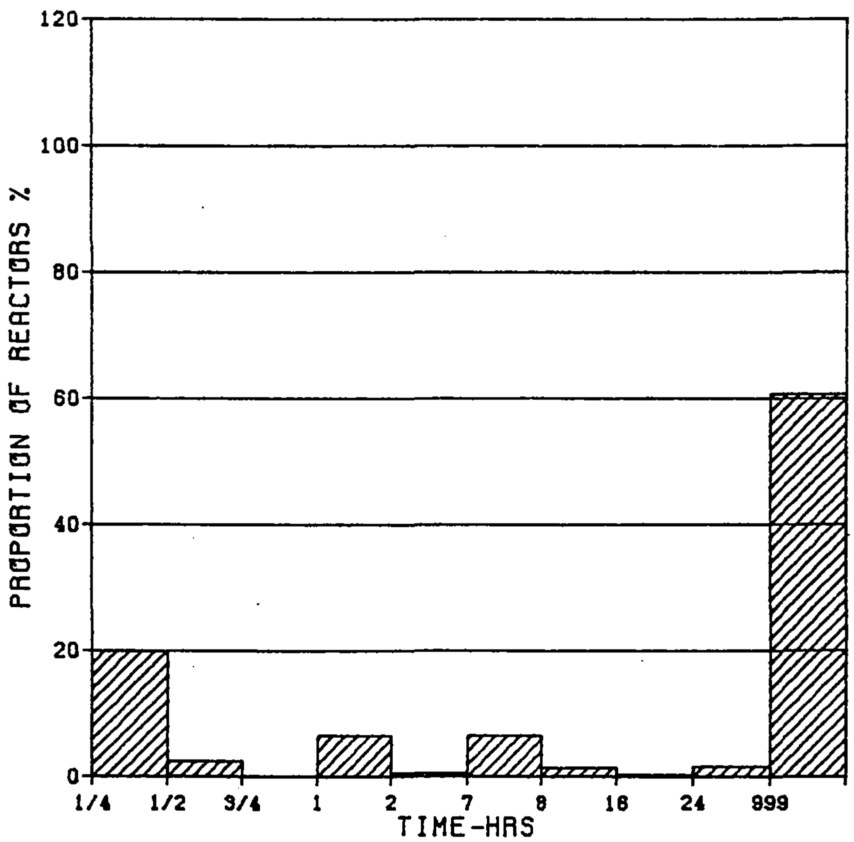
SAMPLE OF REPLIES = 97.9%

SOURCE CIGRE SC 39-04 : 2/5/85

CAPABILITY OF TTH

NUMBER OF REACTORS IN SURVEY : 366

FIG. 44



NOTE: TIMES STATED INDICATE MAXIMUM VALUE
FOR CONTINUED OPERATION ON HOUSE LOAD.

SAMPLE OF REPLIES = 98.1%

SOURCE CIGRE SC 39-04 : 2/5/85

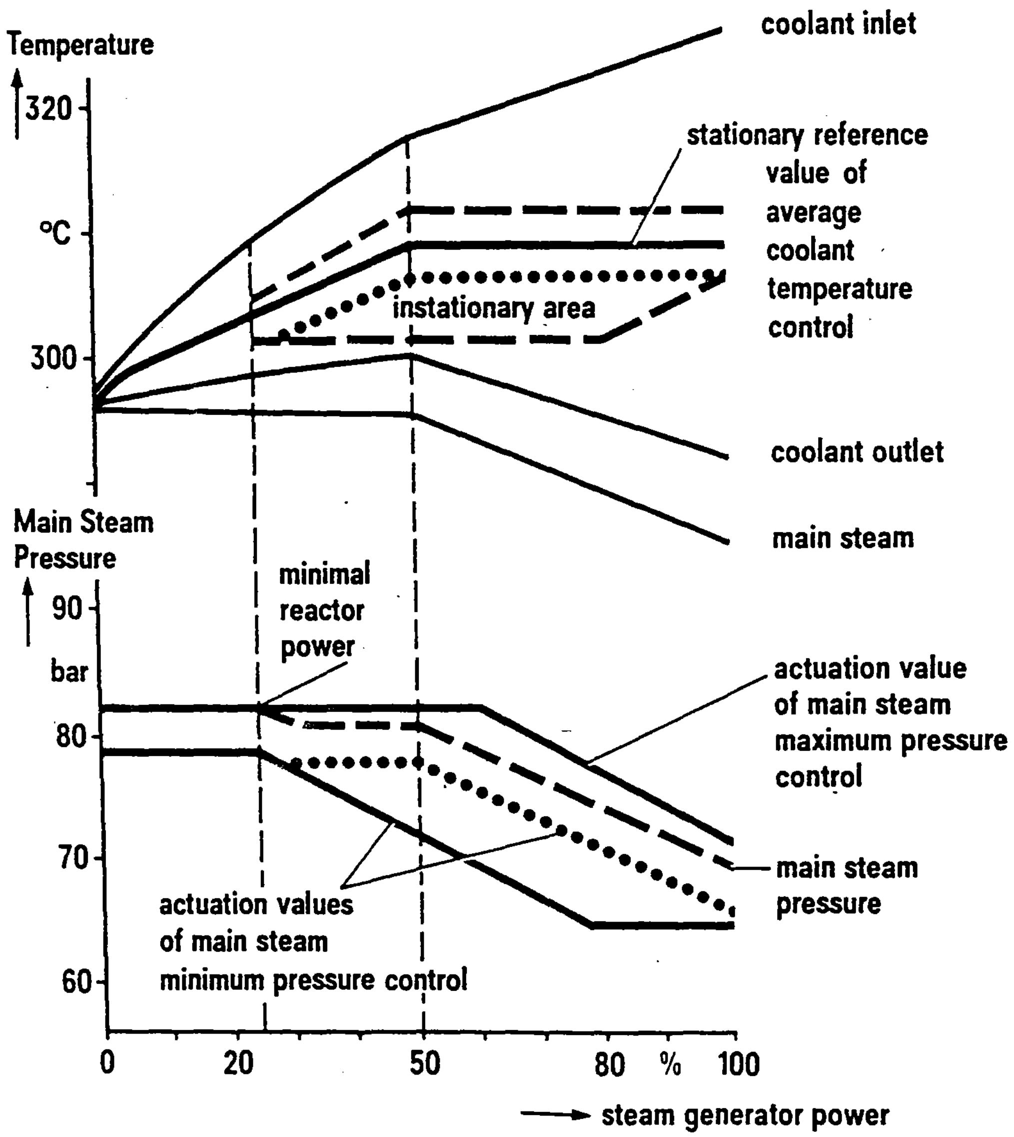
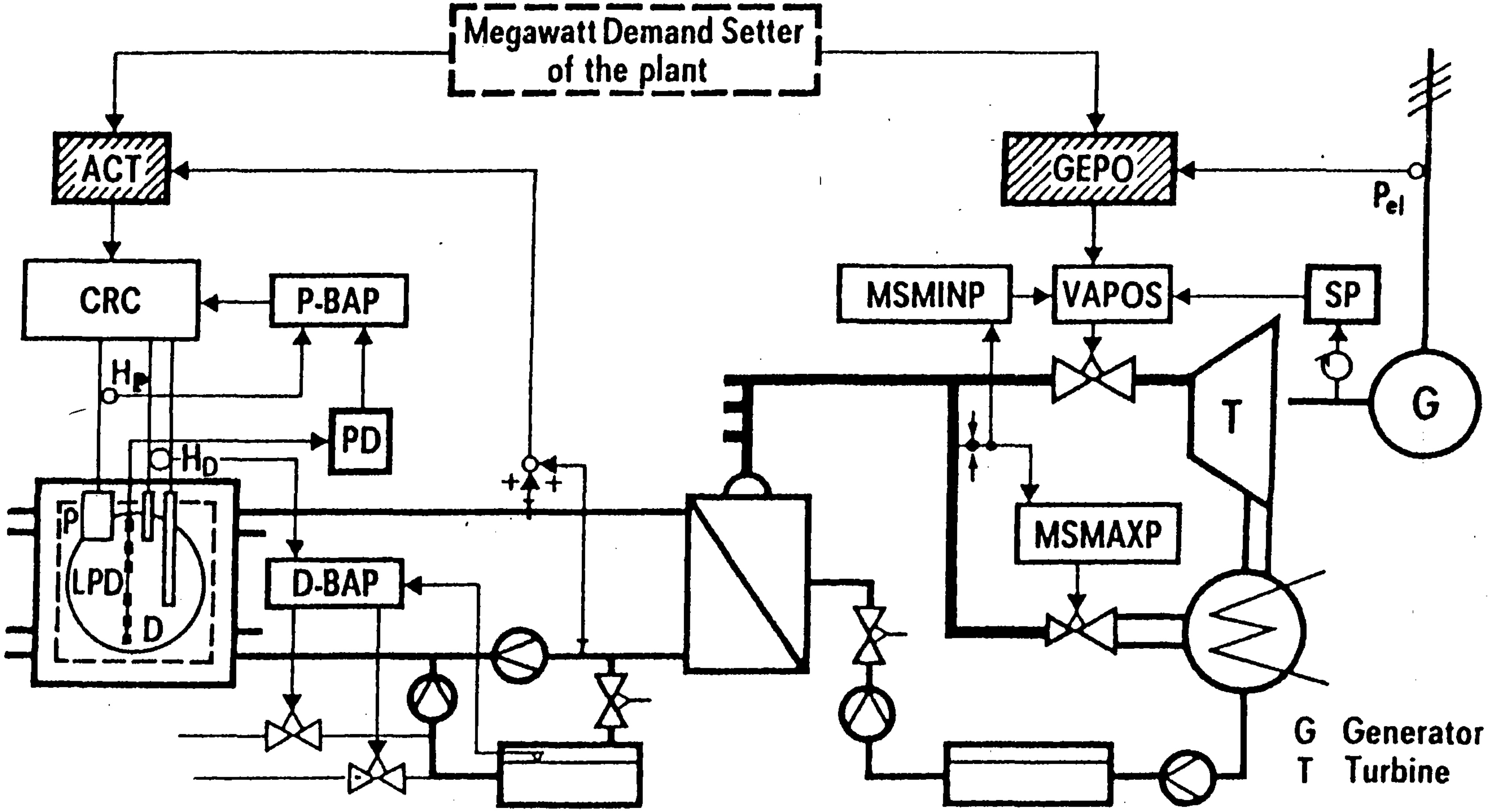


FIGURE 45 STEADY STATE CHARACTERISTICS OF A KWU – PWR

Courtesy – KWU



Feedback controls of:

ACT	Average coolant temperature
P-BAP	P-bank position
D-BAP	D-bank position
PD	Power distribution

CRC	Control rod control
P	Power control rod bank
D	Doppler control rod bank
LPD	Local power detector
P_{el}	Electrical power

Feedback controls of:

GEPO	Generator power
VAPOS	Valve position
SP	Speed
MSMINP	Main steam minimum pressure
MSMAXP	Main steam maximum pressure

FIGURE 46 CONTROL SYSTEM FOR A KWU – PWR (GRAFENRHEINFELD)

Courtesy — KWU

<u>PART 2 : A COMPILATION AND ANALYSIS OF AN INTERNATIONAL SURVEY</u>

* The list of Figures and Tables is included in Part 1 of the Paper with the Full Index at the front - pages (v) and (vi).

1. PREAMBLE

This survey of nuclear power plant design and current and future operational practice in 21 countries was undertaken by a Sub-group of Working Group 04 of CIGRE Study Committee 39 (initially SC32). There is representation from various countries on WG04, by direct participation or by correspondence. This study was initiated in spring 1982 and concluded in early 1985. In considering power system load and frequency control, the Working Group examined international practices in the design provisions for, and the specific rcle of nuclear plant in this context - especially as there was an increasing percentage of nuclear capacity in many countries. This was considered in Part 1.

It was decided to prepare and circulate a questionnaire to utilities in various countries with the WG member, or corresponding member, acting as a focus for replies. In this way, the necessary information could be collected for analysis and presentation. However, in view of the magnitude of the task of data collection, especially for those countries with a large number of utilities, viz USA, it seemed appropriate to consider only a sample of suppliers and operators of Nuclear Steam Supply Systems.

The survey covers 373 power reactors (each of more than 50 MW output) having a total gross generation capacity of 297 GW (282 GWso). Of these, 290 reactors were in service at mid-1985 having a gross capacity of 214 GW (203 GWso). It also includes 83 reactors which are in the construction phase with an expectation of commercial operation by 1990 having a gross capacity of 83 GW (79 GWso). Thus the ratio of operational reactors to the overall sample is 78%, and 72% by generation capacity. By contrast, the IAEA reports that at year end 1984 there were 344 reactor units connected to the grids in 26 countries having a generating capacity of ~ 219 GWso. Thus, this survey covers ~ 84% of the world's operating reactors to end 1984 having a generating capacity of 203 GWso or ~ 93% of world operational capacity.

The main objective of the survey was to examine a representative sample of the designs of NSSS's, from a control and performance viewpoint, and to determine the manner in which utilities operate them now and expect to operate them in the future in a changing demand and grid environment.

The survey and its findings has been made possible by the ready co-operation of utilities, operators, manufacturers, and consultants who contributed substantially in making available the data upon which the survey is based. Those contributing in this way together with relevant data sources are listed in the next section.

2. RESPONDING UTILITIES, MANUFACTURERS, CONSULTANTS and DATA SOURCES *

<u>Argentina</u> CNEA, AECL, KWU

<u>Belgium</u> EBES, INTERCOM

<u>Brazil</u> NUCLEBRAS, ELECTROBRAS

<u>Canada</u> Ontario Hydro, AECL

<u>Finland</u> Imatran Voima OY, TVO

<u>France</u> Electricite de France

Germany	KWO, HEW, NWK, RWE, Neckarwerke, PREAG, Bayernwerke, IAW, Badenwerke EVS, and Kraftwerk Union
Holland	KEMA, PZEM
Hungary	EROTERV
Italy	ENEL
Japan	CRIEPI - Electric Power Industry in Japan, 1984 (Japan Electric Power Information Center, Inc).
Korea	KEPCo, AECL
South Africa	ESCOM
Spain	Hidroelectrica Espanola SA CD de Almaraz
Sweden	The Swedish State Power Board, OKG, Sydkraft
Switzerland	Brown-Boveri, KK Goesgen
Taiwan	Taiwan Power Company
United Kingdom	CEGB, SSEB, Elec. Times Handbook, UKAEA, BNFL
United States of America	Niagara Mohawk Power Corpn., Commonwealth Edison, Pennsylvania Power & Light, GPU Nuclear, Detroit Edison, Illinois Power Co., Portland General Electric, Florida Power Corporation, Louisiana Power & Light, KG&E, KCPL, KEPCo., UE, Indiana & Michigan Electric Co., Toledo Edison Arkansas Power and Light Co., Florida Power & Light Co., Public Service Elect. & Gas Co., Delmarva Power & Light Co., Philadelphia Elec. Co., Atlantic City Elec. Co., Duquesne Light Co., Tennessee Valley Authority and Power Technologies Incorporated.
USSR	References [1,2,3,9,10,11,21,31,68]

3. THE QUESTIONNAIRE

The questionnaire was comprised of 22 questions (0) - (21), with additional space for 'comments' (22) supported by notes of guidance and an example to illustrate the form of replies expected. A reproduction of the questionnaire, together with the example is included in the database as Appendix A5.

4. ANALYSIS OF THE QUESTIONNAIRE REPLIES

In order to provide an efficient presentation, analysis, editing, updating and correction facility for this Survey it was decided at an early stage to establish a computer database of commercial nuclear power plants at the Scientific Services Centre, CEGB - Canal Road, Gravesend, Kent.

* For abbreviations refer [2]

Data was requested for each of 21 countries via WG 04 or its corresponding members. Where this was not productive, recourse was made to published reports and these sources are referenced accordingly. In the case of some countries not represented on the WG, direct correspondence was entered into with the NSSS operating authority. The analysis which follows is based on a compilation of the questionnaire replies. The replies have been embodied as entries to the database file. The database format and presentation is reported fully in the APPENDIX - (Gold Pages).

4.1 Categorization of Power Reactors by output and type

(DATABASE SECTION A1 REFERS)

FIGURE 2* and TABLE 1* list the countries participating in the survey, and show the relative contribution of commissioned nuclear capacity (GWso and %) and installed capacity (GWso) for the start of each of the years 1970, 1980 and a forecast for 1990.

The growth rates for the period to 1990 are greatest for France, Belgium, Japan, FRG, USA and the USSR, whilst the proportion of nuclear capacity to the system capacity > 20% is forecast for France, Belgium, Germany, Finland, Korea, Taiwan and Sweden with the highest percentages applying to Belgium (~ 35%) and France (~ 53%). In practice, of course, the % nuclear contribution by way of generation to each national grid at any one time will vary depending on system load and nuclear plant availability. There will also be a bias on operating cost to the advantage of nuclear relative to fossil generation.

APPENDIX A1 : Pages 1-8 lists the NSSS data for the main plant items, including reactor types comprising the national sample for each of 21 countries. In most cases the sample is 100% for that country. The data of APPENDIX A1 has been analysed, and FIGURES 27 and 28 show the results in the form of pie-charts. The generated output for the overall sample is 297 GW gross (282 GWso) within a family of 373 reactors, of which 85% are Water Reactors which contribute ~ 93% of the overall capacity. World projections for end 1990 are 509 reactors in 32 countries having a total capacity of ~ 370 GWso [2] and [9,2)]. The Figures also show the preponderance of the LWRs in the overall family. The greatest proportion of the capacity is contributed by the PWR (~ 61%) followed by the BWR (~ 19%). Including the LWGR in the BWR category increases the figure for BWRs to 26%. The balance is made up of HWRs (6%) GCRs (6%). Balance, FBRs (1%).

FIGURE 29 shows the mix of plant in the survey both operational and in-construction on a 1985 basis, with a breakdown by reactor type. Within the Survey 78% of the reactors are operational and account for 214 GW gross (203 GWso), or 72% of the capacity which is slated for being in full commercial operation by 1990.

4.2 Contribution of Nuclear Plant to Primary Control

(DATABASE SECTION A2 REFERS)

(a) Types of turbine governing mode

Most plants were governed in the HP mode, but where IP steam admission valves were provided (nearly all the sample) IP governing was, in most cases, also provided. The replies were not included in the database.

* FIGURES and TABLES are at the end of Part 1 of the Paper (WHITE pages)

(b) Regulating Band normally exercised by primary control

FIGURE 30 gives an indication of the extent of 'base-load' regime of operation for nuclear plant at present. It indicates that within the sample of ~ 76% of the total for which replies were received, only ~ 20% of plant operates within a regulating band which is within 10% MCR. Rather more than half the plants do not contribute to primary control. The balance of ~ 30% which includes non-operational plant were 'unknown'.

Discussion of these findings in the WG revealed a need for further work to clarify the specific role of nuclear plant in this respect which might vary according to system loading, in some countries. Thus, at times of light system loading nuclear plant may be required to operate below full load bringing them into a 'regulating band' regime. This is to be covered by further work on the topic of load-cycling of nuclear plant referred in Section 4.3(b).

(c) Average speed droop

The overall droop, i.e. turbine speed change between no-load and full-load, is expressed in APPENDIX A2 as minimum and maximum values, as % MCR^{-1}. FIGURE 31 shows the average value for ~ 81% sample of nuclear plant, this being the arithmetic mean of the values stated. The majority of plants, ~ 83%, have an average governor droop of between 2.5 and 5% MCR^{-1}.

(d) Primary Speed Control Deadband

FIGURE 32 shows the design provisions for deadband in the primary, or speed control loop - see FIGURE 7, speed governor (i). The available replies which are non-zero constitute a small sample, ~ 17% of the overall, indicating that the majority of plants, on a reactor-basis, do not utilize deadband in the turbine speed control loops. Where they exist the design provisions for deadband are in the range ± (50 - 500) mHz.

(e) Normal setting of valve limit

Inspection of APPENDIX A2 reveals that nearly all the nuclear units operate to fully open steam valves. In a few cases this is not so, the limits are usually set by constraints specific to those plants.

4.3 Contribution of Nuclear Plant to Secondary Control

(DATABASE SECTION A.3 REFERS)

(a) Design provisions and control philosophies

FIGURE 33 shows the provisions for secondary control for a sample of reactors comprising ~ 98% of the total. It shows that nearly all the water reactors within the sample include design provisions for secondary control. In the GCR family, the early reactors (Magnox) had no such provisions in any of the listed countries, whereas the later designs (AGR) include automatic facilities for load control ~ 7% of the sample.

FIGURE 34 shows the control philosophy adopted for the reactors, whether RFT, TFR or CC as illustrated in FIGURE 7 with a further sub-division by reactor type to which the philosophy applies. The sample for which replies were received is representative, being ~ 90% of the total. Nearly half the sample incorporate RFT, with smaller groupings for TFR and CC. Some reactors, ~ 25% have more than one mode e.g. the Russian VVER family is capable of TFR mode for base-load duty or RFT for load-following. The strategy is selectable depending upon the System operation requirement [31].

(b) Utilisation - now and in the future

FIGURES 35 and 36 shows the use being made of secondary control in the operation of nuclear plant by reactor type 'now' and in the 'future'. The percentage sample of the whole is good (77 and 72%) but is further improved by the fact that the number of operating reactors is fewer than the overall which includes uncommissioned nuclear plant.

The two pie-charts reflecting the position 'now' and in the 'future' show clearly a changing role for nuclear plant away from base load in the mid 1980s (now) to one of load-frequency control towards the end of the century (future) for some countries. FIGURE 35 shows that within the sample of 77% of the reactors surveyed, only 6% participate in secondary control now (1985 basis), leaving the great majority on base-load. Prospect for the future (1990 basis) as shown in FIGURE 36 indicates just under 50% expecting to load-follow and just over 50% continuing to be base-loaded for a sample of 72% # of the plant surveyed. This trend will influence the utilisation of nuclear plant and their control systems for secondary control as provided in the design stage, and influence control system refurbishment policy for some of the older nuclear plant in some countries - refer also to Part 1 : Section 8.

In considering the questionnaire replies to (16) the Working Group discussed the matter in more detail. The subject of load cycling of nuclear power plant by utilities is becoming increasingly important in some countries. A separate questionnaire and follow-up study is in progress on this topic.*

(c) Auto-control or control range and normal loading rate.

FIGURE 37 shows the results of the analysis of the data. The bar chart indicates the number of reactors with a minimum load value as shown. Replies received for ~ 96% of the reactor sample indicate that, of these, ~ 60% have a minimum load capability of 20% or less. In terms of range and minimum load capability, nuclear power plants compare favourably with fossil-fired plant (oil and gas) and have a wider range than most coal-fired plant.

FIGURE 38 shows the normal value of the plant loading rate as given in APPENDIX A3. About 60% of a 95% sample have a loading rate of 4-5% MCR m^{-1}. Data on loading rate can be subject to many qualifications, some of which relate to NSSS type and others to operating regime, amplitude of change, and method of implementation of the change. Part 1 - Section 4.3 discusses this and FIGURES 9 and 10 show the results of tests carried out by KWU on PWRs in the FRG.

Balance of plant, in construction - see FIGURE 29.

* A recommendation to this effect was agreed by the meeting of SC39 held in Paris on 5 September 1984.

(d) Approximate shape of power response

The replies received against this item illustrated the differences between the various reactor types and NSSSs. To enable meaningful comparisons to be made between the various systems the WG considered that it would be better to illustrate typical responses of NSSS by reactor type for actual incidents, e.g. power profile for a change in demanded load, loss of coolant pump, tripping to houseload, etc. These characteristics are considered in Part 1 - Section 5.2, and FIGURES 9, 13-17.

(e) Daily load-following capability minimum to maximum

The bar chart of FIGURE 37 also shows the design provisions for minimum/maximum load capability for a ~ 96% sample of nuclear power plants. Note however that this diagram does not include the implications of poisoning-out due to extended periods of operation at low load. Data was not available from the replies to suggest whether operational experience permitted the exercising of this capability for load-following. For some countries with a high proportion of nuclear plant, e.g. France, load following by nuclear plant is in operation now. These matters are to be the subject of the follow-up study referred to earlier.

4.4 Capability of nuclear plants for isolated operation

(DATABASE SECTION A4 REFERS)

(a) Turbine Bypass Capacity

FIGURE 39 shows the analysis of replies on the capacity of turbine bypasses to the condenser (database Sections A1 and A4). Within a sample of ~ 97% of the survey, nearly all plants have some provisions for steam bypassing of the turbine to the condenser. No definitive pattern emerges as to such provisions, and examination of the database reveals a variety of design practices even within a given reactor family. Over 50% have capacities between 80-105% MCR steam flow and ~ 30% have capacities between 10-40%. Perhaps the specification of such bypasses depends solely on the operational role anticipated for the plant and its capability and propensity for isolated or islanded operation.

Absence of such facilities, however, does not imply constraints in respect of capability for isolated or islanded operation as discussed in Part 1 - Section 3.7. It is also demonstrated by the incident of islanding and subsequent reconnection and reloading of a 1000 MW nuclear plant which has provisions for steam dumping to atmosphere only, and no provision for live steam dumping to the condenser - ref. Part 1 - Section 6.2(i).

(b) Bypass Control for Start/Shutdown

The replies received indicate that where steam bypasses to the condenser are provided, the start-up/shutdown procedures involve use of these bypasses - refer database entries A4.

(c) Bypass Control for load rejection

In general, the replies indicate that for the great majority of cases, bypasses used in the startup/shutdown procedures are also used during load rejection events. Exceptions occur in the GCR family - refer data base entries A.4.

FIGURE 40 shows the range of houseload for the full reactor sample. Data where it was not available for the USSR designs was taken to be ~ 5% of generated output. This corresponds with the average value for the whole sample.

(d) Isolated operation

FIGURE 41 shows the capability for isolated operation for a large part of the sample of NPPs, ~ 97%, categorized by reactor type. As indicated in Part 1 - Section 4.4 : Provisions for islanding of nuclear plant - several factors influence such provisions. However, the pie-chart indicates that within the sample, ~ 85% of the plant is capable of isolated operation.

(e) Capability of TTH, residual load, and time period.

FIGURE 42 shows the replies received for ~ 97% sample of plants in the survey. Of these about 17% would be expected to shutdown on load rejection. The remainder have a capability of sustained operation on houseload irrespective of steam bypass provisions whether to atmosphere or condensers. The figure also shows the categorization by reactor type.

FIGURE 43 shows the percentage of reactors in the sample which lie within the stated spectrum of % residual electrical load on the generator busbars on a 'per-reactor' basis. The peaking at the low load end suggest that the great proportion of the reactor sample, ~ 85% can maintain operation following load rejection to low loads ≈ Houseload.

FIGURE 44 shows the spread of dwell-times at the residual load for the sample. Those reactors which fall between ¼ hr and ½ hr can be assumed to have only a limited capability for handling TTH, and therefore can be assumed pessimistically to imply a reactor trip condition consequent upon a loss of grid connection. This would still leave ~ 80% capable of continuing in service for more than ½ hr pending a grid reconnection.

5. ACKNOWLEDGEMENTS

The preparation and compilation of this paper has been made possible by the co-operation and support of a number of utilities, manufacturers and consultants, and Working Group members and corresponding members of SC39-04, and associated CIGRE committees. The Sub-Group wish to record their thanks to those who have contributed in this way.

The work and contribution of the Systems Analysis Team at Gravesend responsible for the database is acknowledged, as is the support of the secretarial, Drawing Office, photographic units, and CEGB Translations Section.

The author also wishes to acknowledge the help of utilities in providing data and reports on the grid events referred to and summarised in this paper, including the test results and records for specific events.

APPENDIX

Contents

SPECIAL NOTE

A 'First Draft' of the database dated 26 July 1984 was compiled from data based on the first response to our enquiries. It was circulated to all participating utilities, manufacturers and consultants by August 1984 with an invitation to comment. A draft document containing the 'First Draft' database was presented for discussion at the meeting of WG 04 held in Paris on 4 September 1984, and an invitation was extended to 'update entries' by end November 1984 (subsequently extended to end February 1985).

The present database includes the latest information available, mostly contributed by delegates to WG 04, or their representatives, but in the case of two countries no replies were forthcoming and accordingly we have relied on the most reliable sources for the information presented.

The information contained in this report has been compiled from a number of different sources and it has not been possible to verify its accuracy in each case. The information is believed to be correct at the time of publication but its accuracy is not guaranteed and any person wishing to verify any of the source data should do so directly with the relevant source.

A.0 THE COMPUTER DATABASE

The Database which follows was compiled by members of the staff of the Computer Unit of the Scientific Services Centre of the Central Electricity Generating Board, South Eastern Region, Gravesend, Kent, ENGLAND.

The Database comprises

(i) A DATASET consisting of the replies to the questionnaire enquiry issued to the participating utilities, manufacturers, and consultants. The dataset is stored in the Computer and is not reproduced here as a set of source documents.

(ii) Four TABULAR FORMATS numbered A1 - 4.

These represent the compilation of replies in an ordered form to enable comparisons to be made by direct read-off.

The arrangement of these formats is described below

(iii) Ten PIE CHARTS and eight BAR CHARTS.

These present the results of analysis of the data as contained in the Tabular Formats - (ii). They are the subject of comment in Part 2 and are summarised in Part 1 - Section 9.

THE TABULAR FORMATS

A.1 NSSS DATA FOR 21 COUNTRIES AND CATEGORIZATION OF REACTOR TYPE

	(Reply to Question)
Country	(0)
Reactor Type	(5)a
Utility	(1)
Station	(2)
Year Commissioned	(3)
Total MW Generated	(4)a
House Load	calculated
Number of Reactors	(5)b
Number of Turbines/Reactor	(6)a
Provision of Steam Reheat	(6)b
Turbine By Pass Capacity (to condenser)	(7)

The information presented in Table A1 is by Country, in alphabetic order, and by reactor commissioning dates in time sequence, the earliest reactors being listed first. When no replies were received to the question indicated the entry reads *** (numerical items) and ? (text). This applies to all tables.

A.2 SPEED CONTROL CHARACTERISTICS BY SENSITIVITY AND DEADBAND

	(Reply to Question)
Reactor type	(5)a
Station	(2)
Speed Control Deadband	(11)
Normal Valve Limit	(13)
Regulating Band	(14)
Minimum Droop	(12)a
Maximum Droop	(12)b

The information presented in A.2 is by 'reactor type' grouped together as a reactor family (alphabetically) e.g. BWR, FBR, etc......... PWR, and by Station name, alphabetically.

A.3 LOAD CONTROL - RANGE AND FLEXIBILITY

	(Reply to Question)
Reactor type	(5)a
Station	(2)
Year Commissioned	(3)
Load Control (Secondary)	
(a) Is it provided	(15)a
(b) Philosophy (Figure 7)	(15)b
(c) Utilization - now	(16)a
(d) Utilization - future	(16)b
Auto-control or control range	
(a) Minimum value	(17)a
(b) Maximum value	(17)b
(c) Average loading rate	(17)c

The order of presentation of information in A.3 is similar to that in A.2.

A.4 LOAD REJECTION AND TURBINE BYPASS CONTROL, ISOLATED OPERATION AND PROVISION FOR TTH.

	(Reply to Question)
Reactor type	(5)a
Station	(2)
By Pass (to Condenser) Capacity	(7)
By Pass (to Condenser) Operation during	
(a) Startup and Shut down	(8)
(b) Load Rejection	(9)
Capability of operation without a grid connection	(20)
Provision for TTH	(21)a
Residual Load	(21)b
Time Period	(21)c

The order of presentation of information is as for Table A.2.

THE NUCLEAR PLANT DATABASE - 1985

A.1 LISTING OF NSSS DATA FOR 21 COUNTRIES AND CATEGORIZATION BY REACTOR TYPES

A.2 LISTING OF SPEED CONTROL CHARACTERISTICS BY SENSITIVITY AND DEADBAND

A.3 LISTING OF LOAD CONTROL - RANGE AND FLEXIBILITY

A.4 LISTING OF LOAD REJECTION AND TURBINE BY-PASS CAPACITY, ISOLATED OPERATION AND PROVISIONS FOR TTH.

A1. NSSS DATA FOR 21 COUNTRIES AND CATEGORISATION OF REACTOR TYPE - 2 MAY 1985

NOTE : Y = Yes, N = No, '***' and '?' = Data not provided in questionnaire reply, (*** = numerical, ? = text)

COUNTRY	REACTOR TYPE	UTILITY	STATION	YEAR COM- MISSIONED (ØØ)	TOTAL MW GENERATED	HOUSE LOAD (% MCR)	NUMBER REACTORS	TURBINES /REACTOR	RE- HEAT	TURBINE BYPASS CAPACITY %MCR TO CONDENSER
ARGENTINA	HWR-PV	CNEA	ATUCHA 1	1974	367	5.99	1	1	N	65
	HWR-PT	CNEA	EMBALSE	1983	648	7.41	1	1	Y	70
	HWR-PV	CNEA	ATUCHA 2	1987	745	7.11	1	1	N	80
				TOTAL	1760 ....(1637 MWso)		3			
BELGIUM	PWR	EBES	DOEL 1-2	1975	820	4.39	2	1	Y	85
	PWR	INTERCOM	TIHANGE-1	1975	920	5.43	1	1	Y	85
	PWR	EBES	DOEL-3	1982	936	4.17	1	1	Y	85
	PWR	INTERCOM	TIHANGE-2	1983	941	4.14	1	1	Y	85
	PWR	EBES	DOEL-4	1985	1059	5.00	1	1	Y	85
	PWR	INTERCOM	TIHANGE-3	1985	1048	4.01	1	1	Y	85
				TOTAL	5724 ....(5465 MWso)		7			
BRAZIL	PWR	FURNAS	ANGRA D REIS 1	1982	657	4.72	1	1	Y	85
	PWR	FURNAS	ANGRA D REIS 2,3	1990	2618	4.89	2	1	Y	85
				TOTAL	3275....(3116 MWso)		3			
CANADA	HWR-PT	ONTARIO HYDRO	DOUGLAS PT.	1968	218	5.50	1	1	N	0
	HWR-PT	ONTARIO HYDRO	PICK-A 1-4	1973	2160	4.63	4	1	Y	10
	HWR-PT	ONTARIO HYDRO	BRUCE A 1-4	1978	3200	6.25	4	1	Y	70
	HWR-PT	HYDRO QUEBEC	GENTILLY 2	1982	650	7.69	1	1	Y	70
	HWR-PT	NEW BRUNS. POWER	POINT LEPREAU	1982	650	7.69	1	1	Y	70
	HWR-PT	ONTARIO HYDRO	PICK-B 5-8	1985	2160	4.63	4	1	Y	10
	HWR-PT	ONTARIO HYDRO	BRUCE B 5-8	1987	3200	6.25	4	1	Y	70
	HWR-PT	ONTARIO HYDRO	DARL. A 1-2	1988	1870	5.78	2	1	Y	70
				TOTAL	14108 ...(13288 MWso)		21			
FINLAND	PWR	IMATRAN VOIMA OY	LOVIISA 1,2	1980	930	4.30	2	2	Y	70
	BWR	TEOLLISUUDEN VO	OLKILUOTO 1,2	1982	1470	3.40	2	1	Y	100
				TOTAL	2400 ....(2310 MWso)		4			
FRANCE	GCR	EDF	† CHINON 2	1964	230	13.04	1	2	Y	30
	GCR	EDF	CHINON 3	1966	500	4.00	1	2	Y	30
	GCR-HWR	EDF/CEA	† MONTS D'ARREE	1966	75	6.67	1	*	?	***
	PWR	EDF	CHOOZ (SENA)	1967	320	4.69	1	1	N	0
	GCR	EDF	S LAURENT A1-2	1971	1041	2.59	2	2	?	***
	GCR	EDF/CEA	BUGEY 1	1972	560	2.68	1	2	N	30
	FBR	EDF/CEA	MARCOULE PHENIX	1973	250	6.80	1	2	N	100

† and ØØ - see footnotes on Page 8, Appendix A1.

A1. NSSS DATA FOR 21 COUNTRIES AND CATEGORISATION OF REACTOR TYPE - 2 MAY 1985

NOTE : Y = Yes, N = No, '***' and '?' = Data not provided in questionnaire reply, (*** = numerical, ? = text)

COUNTRY	REACTOR TYPE	UTILITY	STATION	YEAR COM-MISSIONED	TOTAL MW GENERATED	HOUSE LOAD (% MCR)	NUMBER REACTORS	TURBINES /REACTOR	RE-HEAT	TURBINE BYPASS CAPACITY %MCR TO CONDENSER
	PWR	EDF	BUGEY 2-3	1978	1914	3.34	2	1	N	85
	PWR	EDF	FESSENHEIM 1-2	1978	1815	3.03	2	1	N	85
	PWR	EDF	BUGEY 4-5	1979	1866	4.07	2	1	N	85
	PWR	EDF	GRAVELINES B1-4	1981	3804	4.31	4	1	N	85
	PWR	EDF	S LAURENT B1-2	1981	1838	5.33	2	1	N	85
	PWR	EDF	CHINON B1-2	1982	1838	5.33	2	1	N	85
	PWR	EDF	DAMPIERRE 1-4	1982	3748	5.02	4	1	N	85
	PWR	EDF	TRICASTIN 1-4	1982	3820	4.19	4	1	N	85
	PWR	EDF	LE BLAYAIS 1-4	1983	3804	4.31	4	1	N	85
	PWR	EDF	CRUAS 1-4	1984	3684	4.45	4	1	N	85
	PWR	EDF	PALUEL 1-2	1984	2688	4.02	2	1	N	85
	PWR	EDF	FLAMANVILLE 1-2	1985	2688	4.02	2	1	N	85
	PWR	EDF	GRAVELINES C5-6	1985	1902	4.31	2	1	N	85
	PWR	EDF	PALUEL 3-4	1985	2688	4.02	2	1	N	85
	PWR	EDF	S ALBAN 1-2	1985	2696	3.56	2	1	N	85
	PWR	EDF	CATTENOM 1-2	1986	2658	4.82	2	1	N	85
	FBR	NERSA	CREYS-MALVILLE	1986	1240	3.23	1	2	Y	50
	PWR	EDF	BELLEVILLE 1-2	1987	2660	4.14	2	1	N	85
	PWR	EDF	CHINON B3-4	1987	1838	5.33	2	1	N	85
	PWR	EDF	NOGENT 1-2	1988	2660	4.14	2	1	N	85
	PWR	EDF	CATTENOM 3-4	1989	2658	4.82	2	1	N	85
	PWR	EDF	PENLY 1-2	1989	2688	4.02	2	1	N	85
	PWR	EDF	CHOOZ B1	1990	1330	4.51	1	1	N	85
	PWR	EDF	GOLFECH 1	1990	1330	4.14	1	1	N	85
				TOTAL	62831....(59343 MWso)		63			
FRG	PWR	KWO	OBRIGHEIM	1968	345	4.93	1	1	Y	80
	BWR	PREAG	WUERGASSEN	1971	670	4.48	1	1	N	100
	PWR	HEW/NWK	STADE	1972	662	4.83	1	1	Y	80
	PWR	RWE	BIBLIS A	1974	1204	4.73	1	1	Y	45
	PWR	RWE	BIBLIS B	1976	1300	4.77	1	1	Y	45
	BWR	KKB	BRUNSBUETTEL	1976	806	4.34	1	1	Y	90
	PWR	GKN	GKN 1	1976	855	5.26	1	1	Y	45
	BWR	BAG/IAW	ISAR 1	1977	907	4.08	1	1	Y	60
	PWR	NWK/PREAG	UNTERWESER	1978	1300	4.77	1	1	Y	45
	BWR	BAD/EVS	PHILIPPSBURG 1	1979	900	4.00	1	1	Y	115
	PWR	BAYERNW.	GRAFENRHEINFELD	1981	1300	5.46	1	1	Y	45
	BWR	KKK	KRUEMMEL	1983	1316	4.26	1	1	Y	80
	PWR	PREAG	GROHNDE	1984	1361	5.36	1	1	Y	45
	BWR	KGB	GUNDREM. 2B 2C	1984	2620	5.04	2	1	Y	60
	PWR	BAD/EVS	PHILIPPSBURG 2	1984	1349	6.00	1	1	Y	45
	HTGR	HKG	UENTROP	1984	307	3.58	1	1	Y	100

A1. NSSS DATA FOR 21 COUNTRIES AND CATEGORISATION OF REACTOR TYPE – 2 MAY 1985

NOTE : Y = Yes, N = No, '***' and '?' = Data not provided in questionnaire reply, (*** = numerical, ? = text)

COUNTRY	REACTOR TYPE	UTILITY	STATION	YEAR COMMISSIONED	TOTAL MW GENERATED	HOUSE LOAD (% MCR)	NUMBER REACTORS	TURBINES /REACTOR	RE-HEAT	TURBINE BYPASS CAPACITY %MCR TO CONDENSER
	PWR	RWE	MUELHEIM-K	1985	1306	6.36	1	1	Y	95
	PWR	NWK/HEW	BROKDORF	1986	1365	5.49	1	1	Y	45
	FBR	SBK	KALKAR	1986	327	9.79	1	1	N	100
	PWR	VEW	EMSLAND	1988	1301	4.53	1	1	Y	45
	PWR	BAG/IAW	ISAR 2	1988	1350	5.93	1	1	Y	45
	PWR	GKN	GKN 2	1989	1314	6.39	1	1	Y	45
				TOTAL	24165....(22915 MWso)		23			
GB †	GCR	BNFL	CALDER HALL	1956	240	25.00	4	2	N	100
	GCR	BNFL	CHAPELCROSS	1960	240	20.00	4	2	N	100
	GCR	CEGB	BERKELEY	1962	332	16.87	2	2	N	20
	GCR	CEGB	BRADWELL	1962	312	3.85	2	3	N	20
	GCR	SSEB	HUNTERSTON A	1964	360	16.67	2	3	N	20
	GCR	CEGB	DUNGENESS A	1965	570	3.51	2	2	N	20
	GCR	CEGB	HINKLEY.PT A	1965	560	10.71	2	3	N	20
	GCR	CEGB	TRAWSFYNYDD	1965	580	13.79	2	2	N	20
	GCR	CEGB	SIZEWELL	1966	650	10.77	2	1	Y	20
	GCR	CEGB	OLDBURY	1967	626	4.15	2	1	Y	20
	BWR-PT	UKAEA	WINFRITH	1968	100	8.00	1	1	N	40
	GCR	CEGB	WYLFA	1971	1340	11.94	2	2	Y	20
	FBR	UKAEA	DOUNREAY PFR	1975	250	8.00	1	1	Y	50
	GCR-AGR	CEGB	HINKLEY.PT B	1976	1330	6.62	2	1	Y	20
	GCR-AGR	SSEB	HUNTERSTON B	1976	1330	7.52	2	1	Y	20
	GCR-AGR	CEGB	DUNGENESS B	1983-5	1320	9.09	2	1	Y	40
	GCR-AGR	CEGB	HARTLEPOOL	1983	1332	6.16	2	1	Y	20
	GCR-AGR	CEGB	HEYSHAM 1	1983	1320	6.06	2	1	Y	20
	GCR-AGR	CEGB	HEYSHAM 2	1988	1330	6.62	2	1	Y	20
	GCR-AGR	SSEB	TORNESS	1988	1320	6.06	2	1	Y	20
				TOTAL	15442....(14124 MWso)		42			
HUNGARY	PWR	MVMT	PAKS 1-2	1984	880	6.36	2	2	Y	100
	PWR	MVMT	PAKS 3-4	1987	880	6.36	2	2	Y	100
				TOTAL	1760....(1648 MWso)		4			
ITALY	GCR	ENEL	LATINA	1964	160	6.25	1	2	N	***
	PWR	ENEL	TRINO VERCL.	1965	272	4.04	1	2	Y	7
	BWR	ENEL	CAORSO	1981	894	2.13	1	1	Y	32
	BWR	ENEL	ALTO LAZIO 1-2	1990	2018	2.68	2	1	N	35
				TOTAL	3344....(3250 MWso)		5			

† MW data for GB represents full plant capability for all reactors

A1. NSSS DATA FOR 21 COUNTRIES AND CATEGORISATION OF REACTOR TYPE - 2 MAY 1985

NOTE : Y = Yes, N = No, '***' and '?' = Data not provided in questionnaire reply, (*** = numerical, ? = text)

COUNTRY	REACTOR TYPE	UTILITY	STATION	YEAR COM-MISSIONED	TOTAL MW GENERATED	HOUSE LOAD (% MCR)	NUMBER REACTORS	TURBINES /REACTOR	RE-HEAT	TURBINE BYPASS CAPACITY %MCR TO CONDENSER
JAPAN	GCR	JAPCO	TOKAI 1	1966	166	9.64	1	2	N	10
	PWR	KANSAI EP CO	MIHAMA 1	1970	340	5.88	1	1	Y	10
	BWR	JAPCO	TSURUGA 1	1970	357	4.76	1	2	N	15
	BWR	TOKYO EP CO	F.DAIICHI 1	1971	460	4.57	1	1	N	105
	PWR	KANSAI EP CO	MIHAMA 2	1972	500	6.00	1	1	Y	40
	BWR	TOKYO EP CO	F.DAIICHI 2	1974	784	3.06	1	1	N	25
	BWR	CHUKOGU EP CO	SHIMANE 1	1974	460	5.00	1	1	N	105
	PWR	KANSAI EP CO	TAKAHAMA 1	1974	826	5.57	1	1	Y	40
	PWR	KYUSHU EP CO	GENKAI 1	1975	559	5.37	1	1	Y	40
	PWR	KANSAI EP CO	TAKAHAMA 2	1975	826	5.57	1	1	Y	40
	BWR	TOKYO EP CO	F.DAIICHI 3	1976	784	3.06	1	1	N	25
	BWR	CHUBU EP CO	HAMAOKA 1	1976	540	4.44	1	1	N	25
	PWR	KANSAI EP CO	MIHAMA 3	1976	826	5.57	1	1	Y	40
	PWR	SHIKOKU EP CO	IKATA 1	1977	566	4.95	1	1	Y	40
	BWR	TOKYO EP CO	F.DAIICHI 4-5	1978	1568	3.06	2	1	N	25
	BWR	CHUBU EP CO	HAMAOKA 2	1978	840	3.10	1	1	N	25
	BWR	JAPCO	TOKAI 2	1978	1160	3.88	1	1	N	25
	BWR	TOKYO EP CO	F.DAIICHI 6	1979	1100	3.00	1	1	N	25
	PWR	KANSAI EP CO	OHI 1-2	1979	2350	4.68	2	1	Y	40
	PWR	KYUSHU EP CO	GENKAI 2	1981	559	5.37	1	1	Y	40
	BWR	TOKYO EP CO	F.DAINI 1	1982	1100	3.00	1	1	N	25
	PWR	SHIKOKU EP CO	IKATA 2	1982	566	4.95	1	1	Y	40
	BWR	TOKYO EP CO	F.DAINI 2	1984	1100	3.00	1	1	N	25
	BWR	TOHOKU EP CO	ONAGAWA 1	1984	524	4.96	1	1	N	25
	PWR	KYUSHU EP CO	SENDAI 1	1984	890	4.94	1	1	Y	40
	BWR	TOKYO EP CO	F.DAINI 3	1985	1100	3.00	1	1	N	100
	BWR	TOKYO EP CO	KASHIWAZAKI-K1	1985	1100	3.00	1	1	N	25
	PWR	KANSAI EP CO	TAKAHAMA 3-4	1985	1740	4.60	2	1	Y	70
	PWR	KYUSHU EP CO	SENDAI 2	1986	890	4.94	1	1	Y	40
	BWR	TOKYO EP CO	F.DAINI 4	1987	1100	3.00	1	1	N	100
	BWR	CHUBU EP CO	HAMAOKA 3	1987	1100	5.00	1	1	Y	25
	PWR	JAPCO	TSURUGA 2	1987	1160	3.88	1	1	Y	90
	BWR	CHUGOKU EP CO	SHIMANE 2	1989	820	4.88	1	1	N	100
	PWR	HOKKAIDO EP CO	TOMARI 1	1989	579	5.01	2	1	Y	70
	PWR	SHIKOKU EP CO	IKATA 3	1990	890	4.94	1	1	Y	40
	BWR	TOKYO EP CO	KASHIWAZAKI-K2-3	1990	2200	3.00	2	1	N	100
	FBR	PNC	MONJU	1990	280	5.36	1	1	Y	50
	PWR	HOKKAIDO EP CO	TOMARI 2	1990	579	5.01	2	1	Y	70
				TOTAL	33289 ....(31892 MWso)		44			
KOREA	PWR	KEPCO	KNU 1	1978	587	5.11	1	1	Y	40
	PWR	KEPCO	KNU 2	1983	650	3.38	1	1	Y	40

A1. NSSS DATA FOR 21 COUNTRIES AND CATEGORISATION OF REACTOR TYPE - 2 MAY 1985

NOTE : Y = Yes, N = No, '***' and '?' = Data not provided in questionnaire reply, (*** = numerical, ? = text)

COUNTRY	REACTOR TYPE	UTILITY	STATION	YEAR COM-MISSIONED	TOTAL MW GENERATED	HOUSE LOAD (% MCR)	NUMBER REACTORS	TURBINES /REACTOR	RE-HEAT	TURBINE BYPASS CAPACITY %MCR TO CONDENSER
	HWR	KEPCO	KNU 3	1983	678	7.37	1	1	Y	90
	PWR	KEPCO	KNU 5	1985	950	5.79	1	1	Y	40
	PWR	KEPCO	KNU 6	1985	950	5.79	1	1	Y	40
	PWR	KEPCO	KNU 7	1986	950	5.26	1	1	Y	40
	PWR	KEPCO	KNU 8	1987	950	5.26	1	1	Y	40
	PWR	KEPCO	KNU 9-10	1989	1900	5.26	2	1	Y	85
				TOTAL	7615 ....(7203 MWso)		9			
NETHERLANDS	BWR	GKN	DODEWAARD	1968	58	5.17	1	1	Y	100
	PWR	PZEM	BORSSELE	1973	481	4.37	1	1	Y	90
				TOTAL	539 ....(515 MWso)		2			
ROMANIA	PWR	(VVER-440)	OLT	1983	440	6.36	1	2	Y	100
	HWR-PT	INC	CERNAVODA	1990	1358	7.36	2	1	Y	70
				TOTAL	1798 ....(1670 MWso)		3			
S. AFRICA	PWR	ESCOM	KOEBERG 1-2	1984	1930	4.56	2	1	Y	90
				TOTAL	1930 ....(1842 MWso)		2			
SPAIN	PWR	UE FENOSA	JOSE CABRERA	1969	160	4.38	1	1	?	***
	BWR	NUCLENOR	SM. GARONA	1971	460	4.35	1	1	?	***
	GCR	HIFRENSA	VANDELLOS 1	1972	496	3.23	1	2	Y	***
	PWR	CSE/HE/UE	ALMARAZ 1	1980	930	3.23	1	1	Y	40
	PWR	CSE/HE/UE	ALMARAZ 2	1983	930	3.23	1	1	Y	40
	BWR	HE	COFRENTES	1984	974	4.41	1	1	Y	35
	PWR	FECSA+	ASCO 1-2	1985	1860	4.62	2	1	Y	40
	PWR	UE/ERZ	TRILLO 1	1987	1157	10.03	1	1	Y	54
	PWR	HIFRENSA	VANDELLOS 2	1987	982	7.13	1	1	Y	***
				TOTAL	7949 ....(7531 MWso)		10			
SWEDEN	BWR	OKG	OSKARSHAMN 1	1972	460	4.35	1	1	Y	100
	BWR	OKG	OSKARSHAMN 2	1975	615	7.32	1	1	Y	100
	PWR	SSPB	RINGHALS 2	1975	840	4.76	1	2	Y	90
	BWR	SSPB	RINGHALS 1	1976	780	3.85	1	2	Y	104
	BWR	SYDKRAFT	BARSEBECK 1-2	1977	1180	3.39	2	1	Y	100
	BWR	SSPB	FORSMARK 1-2	1981	1876	4.05	2	2	Y	100
	PWR	SSPB	RINGHALS 3-4	1983	1920	4.69	2	2	Y	90
	BWR	SSPB	FORSMARK 3	1985	1090	3.67	1	1	Y	100

A1. NSSS DATA FOR 21 COUNTRIES AND CATEGORISATION OF REACTOR TYPE - 2 MAY 1985

NOTE : Y = Yes, N = No, '***' and '?' = Data not provided in questionnaire reply, (*** = numerical, ? = text)

COUNTRY	REACTOR TYPE	UTILITY	STATION	YEAR COMMISSIONED	TOTAL MW GENERATED	HOUSE LOAD (% MCR)	NUMBER REACTORS	TURBINES /REACTOR	RE-HEAT	TURBINE BYPASS CAPACITY %MCR TO CONDENSER
	BWR	OKG	OSKARSHAMN 3	1985	1100	3.64	1	1	Y	100
				TOTAL	9861 ....(9440 MWso)		12			
SWITZERLAND	PWR	NOK AG	BEZNAU 1	1969	364	3.85	1	2	Y	17
	PWR	NOK AG	BEZNAU 2	1971	364	3.85	1	2	Y	17
	BWR	BWK AG	MUHLEBERG	1971	326	4.29	1	2	Y	110
	PWR	KKG	GOESGEN	1979	970	5.15	1	1	Y	45
	BWR	KKL	LEIBSTADT	1984	1000	5.50	1	1	Y	110
				TOTAL	3024 ....(2877 MWso)		5			
TAIWAN	BWR	TPC	CHINSHAN 1	1978	636	5.03	1	1	Y	25
	BWR	TPC	CHINSHAN 2	1979	636	5.03	1	1	Y	25
	BWR	TPC	KUOSHENG 1	1981	985	3.76	1	1	Y	35
	BWR	TPC	KUOSHENG 2	1983	985	3.76	1	1	Y	35
	PWR	TPC	MAANSHAN 1	1984	951	6.41	1	1	Y	32
	PWR	TPC	MAANSHAN 2	1985	951	6.41	1	1	Y	32
				TOTAL	5144 ....(4884 MWso)		6			
USA	BWR	NIAGARA MPC	NINE MILE POINT	1969	630	3.17	1	1	Y	42
	BWR	JERSEY CENTRAL P	OYSTER CREEK 1	1969	670	2.99	1	1	N	40
	BWR	COMMONWEALTH ED	DRESDEN 2-3	1971	1666	4.68	2	1	N	40
	BWR	COMMONWEALTH ED	QUAD CITIES 1-2	1972	1666	5.28	2	1	N	40
	PWR	FLORIDA PLC.	TURKEY PT 3-4	1973	1398	4.72	2	1	Y	50
	PWR	ARKANSAS PLC	ARKANSAS N1-1	1974	883	5.32	1	1	Y	17
	PWR	GPU NUC	TMI-1	1974	837	5.73	1	1	N	21
	PWR	COMMONWEALTH ED	ZION 1-2	1974	2170	4.15	2	1	Y	40
	PWR	INDIANA & MIC	DONALD C COOK 1	1975	1070	3.74	1	1	Y	85
	PWR	PGE/PPL	TROJAN	1975	1178	4.07	1	1	Y	50
	PWR	DUQUESNE LC	BEAVER VALLEY 1	1976	860	5.81	1	1	?	85
	BWR	TVA	BROWNS FERRY 1	1977	3294	2.85	3	1	N	60
	PWR	FLORIDA PC.	CRYSTAL R.3	1977	875	4.46	1	1	Y	15
	PWR	TOLED/CEIC	DAVIS-BESSE 1	1977	928	5.82	1	1	Y	25
	PWR	PSEG/ACE/DPL/PEC	SALEM 1	1977	1090	4.59	1	1	Y	50
	PWR	INDIANA & MIC	DONALD C COOK 2	1978	1140	3.51	1	1	Y	85
	PWR	ARKANSAS PLC	ARKANSAS N1-2	1980	897	4.35	1	1	Y	27
	PWR	PSEG/ACE/DPL/PEC	SALEM 2	1981	1115	4.48	1	1	Y	50
	PWR	TVA	SEQUOYAH 1-2	1981	2366	2.96	2	1	Y	40
	BWR	COMMONWEALTH ED	LA SALLE C 1	1982	1122	3.92	1	1	Y	25
	PWR	FLORIDA PLC.	ST.LUCIE 1-2	1983	1732	5.08	2	1	Y	45
	PWR	UNION EC	CALLAWAY 1	1984	1234	5.11	1	1	Y	40

A1. NSSS DATA FOR 21 COUNTRIES AND CATEGORISATION OF REACTOR TYPE - 2 MAY 1985

NOTE : Y = Yes, N = No, '***' and '?' = Data not provided in questionnaire reply, (*** = numerical, ? = text)

COUNTRY	REACTOR TYPE	UTILITY	STATION	YEAR COM-MISSIONED	TOTAL MW GENERATED	HOUSE LOAD (% MCR)	NUMBER REACTORS	TURBINES /REACTOR	RE-HEAT	TURBINE BYPASS CAPACITY %MCR TO CONDENSER
	BWR	COMMONWEALTH ED	LA SALLE C 2	1984	1122	3.92	1	1	Y	25
	BWR	PENN POWER & LC	SUSQUEHANNA SE	1984	2170	3.23	2	1	N	25
	BWR	DETROIT EDISON	ENRICO FERMI 2	1985	1154	5.29	1	1	Y	27
	PWR	LOUISIANA PLC.	WATERFORD SES 3	1985	1153	4.25	1	1	Y	63
	PWR	KG&E/KCPL/KEPCO	WOLF CREEK	1985	1234	5.11	1	1	Y	40
	PWR	DUQUESNE LC	BEAVER VALLEY 2	1986	860	5.81	1	1	?	85
	BWR	ILLINOIS PC	CLINTON	1986	985	5.28	1	1	?	35
				TOTAL	37499....(35884 MWso)		38			
		(REACTOR TYPE)⊕								
USSR	LWGR	RBMK-100	TROITSK	1958	600	6.00	6	1	Y	100
	LWGR	RBMK-100	BELOYARSK 1	1964	100	6.00	1	1	Y	100
	PWR	VVER-210	NOVOVORONEZH 1	1964	210	5.71	1	2	Y	100
	LWGR	RBMK-200	BELOYARSK 2	1967	200	6.00	1	2	Y	100
	PWR	VVER-365	NOVOVORONEZH 2	1970	365	6.03	1	2	Y	100
	PWR	VVER-440	NOVOVORONEZH 3-4	1973	880	6.36	2	2	Y	100
	FBR	BN-350	SHEVCHENKO	1973	350	4.00	1	3	Y	100
	PWR	VVER-440	KOLA 1-2	1974	880	6.36	2	2	Y	100
	LWGR	RBMK-1000	LENINGRAD 1-2	1975	2000	4.00	2	2	Y	100
	LWGR	RBMK-1000	CHERNOBYL 1-2	1978	2000	4.00	2	2	Y	100
	LWGR	RBMK-1000	KURSK 1-2	1979	2000	4.00	2	2	Y	100
	PWR	VVER-440	ARMYANSK 1-2	1980	880	6.36	2	2	Y	100
	FBR	BN-600	BELOYARSK 3	1981	600	4.00	1	3	Y	100
	LWGR	RBMK-1000	LENINGRAD 3-4	1981	2000	4.00	2	2	Y	100
	PWR	VVER-1000	NOVOVORONEZH 5	1981	1000	4.70	1	2	Y	100
	PWR	VVER-440	ROVNO 1-2	1982	880	6.36	2	2	Y	100
	LWGR	RBMK-1000	KURSK 3	1983	1000	4.00	1	2	Y	100
	LWGR	RBMK-1000	CHERNOBYL 3-4	1984	2000	4.00	2	2	Y	100
	LWGR	RBMK-1500	IGNALINA 1	1984	1500	3.33	1	2	Y	100
	PWR	VVER-1000	KALININ 1-2	1984	2000	4.00	2	2	Y	100
	PWR	VVER-1000	KHMELNITSKI 1	1984	1000	4.00	1	2	Y	100
	PWR	VVER-440	KOLA 3-4	1984	880	6.36	2	2	Y	100
	PWR	VVER-1000	NIKOLAYEV 1	1984	1000	4.00	1	2	Y	100
	LWGR	RBMK-1000	SMOLENSK 1-2	1984	2000	4.00	2	2	Y	100
	PWR	VVER-1000	TSIMLYANSK 1	1984	1000	4.00	1	2	Y	100
	PWR	VVER-1000	ZAPOROZHE 1	1985	1000	4.00	1	2	Y	100
	PWR	VVER-1000	AKTASH 1	1986	1000	4.00	1	2	Y	100
	LWGR	RBMK-1500	IGNALINA 2	1986	1500	3.33	1	2	Y	100
	PWR	VVER-1000	KHMELNITSKI 2	1986	1000	4.00	1	2	Y	100
	LWGR	RBMK-1000	KURSK 4	1986	1000	4.00	1	2	Y	100
	PWR	VVER-1000	NIZHNEKAMSK 1-2	1986	2000	4.00	2	2	Y	100
	PWR	VVER-1000	ROVNO 3	1986	1000	4.00	1	2	Y	100
	PWR	VVER-1000	TSIMLYANSK 2	1986	1000	4.00	1	2	Y	100

A1. NSSS DATA FOR 21 COUNTRIES AND CATEGORISATION OF REACTOR TYPE - 2 MAY 1985

NOTE : Y = Yes, N = No, '***' and '?' = Data not provided in questionnaire reply, (*** = numerical, ? = text)

COUNTRY	REACTOR TYPE	UTILITY	STATION	YEAR COM-MISSIONED	TOTAL MW GENERATED	HOUSE LOAD (% MCR)	NUMBER REACTORS	TURBINES /REACTOR	RE-HEAT	TURBINE BYPASS CAPACITY %MCR TO CONDENSER
		(REACTOR TYPE)								
	LWGR	RBMK-1500	KOSTROMA 1	1987	1500	3.33	1	2	Y	100
	PWR	VVER-1000	NIKOLAYEV 2	1987	1000	4.00	1	2	Y	100
	PWR	VVER-1000	ODESSA 1-2	1987	2000	4.00	2	2	Y	100
	PWR	VVER-1000	AKTASH 2	1988	1000	4.00	1	2	Y	100
	PWR	VVER-1000	NIZHNEKAMSK 3-4	1988	2000	4.00	2	2	Y	100
	PWR	VVER-1000	TATAR 1	1988	1000	4.00	1	2	Y	100
	PWR	VVER-1000	ZAPOROZHE 2-4	1988	3000	4.00	3	2	Y	100
	PWR	VVER-1000	NEFTYKAMSK 1	1989	1000	4.00	1	2	Y	100
	PWR	VVER-1000	KALININ 3-4	1990	2000	4.00	2	2	Y	100
	PWR	VVER-1000	NIKOLAYEV 3-4	1990	2000	4.00	2	2	Y	100
				TOTAL	53325....(51082 MWso)		67			
				TOTAL	296782....(281916 MWso)		373			

Footnotes:

† Reactors which are expected to be decommissioned in 1985.

øø Year commissioned for units, or last unit in that station.

⊕ See Section 3.1 - Part 1.

A2. SPEED CONTROL CHARACTERISTICS BY SENSITIVITY AND DEADBAND - 2 MAY 1985

'***' and '?' = Data not provided in questionnaire reply, (*** = numerical, ? = text)

REACTOR TYPE	STATION	SPEED CONTROL DEADBAND +-mHZ	NORMAL VALVE LIMIT (%)	REGULATING BAND %MCR	DROOP MIN (% MCR)	DROOP MAX (% MCR)
BWR	ALTO LAZIO 1-2	0.00	***	***	3	8
	BARSEBECK 1-2	0.00	50	0	6	6
	BROWNS FERRY 1	0.00	100	0	5	5
	BRUNSBUETTEL	0.00	100	40	2	8
	CAORSO	0.00	100	0	***	***
	CHINSHAN 1	200.00	100	0	5	5
	CHINSHAN 2	200.00	100	0	5	5
	CLINTON	0.00	100	3	5	5
	COFRENTES	0.00	100	12	3	8
	DODEWAARD	*******	100	0	1	10
	DRESDEN 2-3	0.00	100	0	3	7
	ENRICO FERMI 2	0.00	100	0	3	7
	F.DAIICHI 1	250.00	100	0	5	5
	F.DAIICHI 2	250.00	100	0	5	5
	F.DAIICHI 3	250.00	100	0	5	5
	F.DAIICHI 4-5	250.00	100	0	5	5
	F.DAIICHI 6	250.00	100	0	5	5
	F.DAINI 1	250.00	100	0	5	5
	F.DAINI 2	250.00	100	0	5	5
	F.DAINI 3	250.00	100	0	5	5
	F.DAINI 4	250.00	100	0	5	5
	FORSMARK 1-2	0.00	100	0	4	4
	FORSMARK 3	0.00	100	0	4	4
	GUNDREM.2B 2C	0.00	100	0	2	8
	HAMAOKA 1	300.00	100	0	3	5
	HAMAOKA 2	300.00	100	0	3	5
	HAMAOKA 3	300.00	100	0	5	3
	ISAR 1	0.00	100	0	2	8
	KASHIWAZAKI-K1	250.00	100	0	5	5
	KASHIWAZAKI-K2-3	250.00	100	0	5	5
	KRUEMMEL	*******	100	***	2	8
	KUOSHENG 1	200.00	100	0	5	5
	KUOSHENG 2	200.00	100	0	5	5
	LA SALLE C 1	0.00	100	0	3	7
	LA SALLE C 2	0.00	100	0	3	7
	LEIBSTADT	0.00	100	0	2	10
	MUHLEBERG	0.00	100	0	4	4
	NINE MILE POINT	100.00	100	15	***	0
	OLKILUOTO 1,2	0.00	100	5	1	6
	ONAGAWA 1	300.00	37	0	5	5
	OSKARSHAMN 1	0.00	100	0	4	4

A2. SPEED CONTROL CHARACTERISTICS BY SENSITIVITY AND DEADBAND - 2 MAY 1985

'***' and '?' = Data not provided in questionnaire reply, (*** = numerical, ? = text)

REACTOR TYPE	STATION	SPEED CONTROL DEADBAND +-mHZ	NORMAL VALVE LIMIT (%)	REGULATING BAND %MCR	DROOP MIN (% MCR)	DROOP MAX (% MCR)
	OSKARSHAMN 2	0.00	100	0	4	4
	OSKARSHAMN 3	0.00	100	0	4	4
	OYSTER CREEK 1	0.00	100	0	3	7
	PHILIPPSBURG 1	0.00	100	0	2	8
	QUAD CITIES 1-2	0.00	100	0	3	7
	RINGHALS 1	0.00	85	0	4	4
	SHIMANE 1	0.00	100	0	4	4
	SHIMANE 2	0.00	100	0	4	4
	SM. GARONA	*******	100	***	***	***
	SUSQUEHANNA SE	0.00	90	0	5	5
	TOKAI 2	250.00	55	0	5	5
	TSURUGA 1	600.00	60	0	5	5
	WINFRITH	0.00	100	0	4	6
	WUERGASSEN	100.00	100	0	2	8
FBR	BELOYARSK 3	*******	***	***	***	***
	CREYS-MALVILLE	0.00	100	0	4	4
	DOUNREAY PFR	0.00	100	0	4	8
	KALKAR	0.00	100	0	2	8
	MARCOULE PHENIX	0.00	100	0	4	4
	MONJU	0.00	100	0	3	7
	SHEVCHENKO	*******	***	***	***	***
GCR	BERKELEY	0.00	100	0	4	4
	BRADWELL	0.00	100	0	4	4
	BUGEY 1	0.00	100	0	4	4
	CALDER HALL	0.00	100	0	4	4
	CHAPELCROSS	0.00	100	0	4	4
†	CHINON 2	0.00	100	0	4	4
	CHINON 3	0.00	100	0	4	4
	DUNGENESS A	0.00	100	0	4	4
	DUNGENESS B	0.00	100	0	1	4
	HARTLEPOOL	0.00	100	0	4	20
	HEYSHAM 1	0.00	100	0	4	20
	HEYSHAM 2	0.00	100	0	4	20
	HINKLEY.PT A	0.00	100	0	4	4
	HINKLEY.PT B	0.00	100	0	1	20
	HUNTERSTON A	0.00	100	0	4	4
	HUNTERSTON B	0.00	100	0	1	25
	LATINA	*******	100	0	0	0
†	MONTS D'ARREE	0.00	100	0	4	4

A2. SPEED CONTROL CHARACTERISTICS BY SENSITIVITY AND DEADBAND - 2 MAY 1985

'***' and '?' = Data not provided in questionnaire reply, (*** = numerical, ? = text)

REACTOR TYPE	STATION	SPEED CONTROL DEADBAND +-mHZ	NORMAL VALVE LIMIT (%)	REGULATING BAND %MCR	DROOP MIN (% MCR)	DROOP MAX (% MCR)
	OLDBURY	0.00	100	0	4	4
	S LAURENT A1-2	0.00	100	0	4	4
	SIZEWELL	0.00	100	0	4	4
	TOKAI 1	750.00	65	0	6	6
	TORNESS	0.00	100	0	4	20
	TRAWSFYNYDD	0.00	100	0	4	4
	VANDELLOS 1	*******	100	0	***	***
	WYLFA	0.00	100	0	4	4
HTGR	UENTROP	20.00	100	0	3	6
HWR	ATUCHA 1	500.00	100	***	3	8
	ATUCHA 2	500.00	100	0	3	8
	BRUCE A 1-4	0.00	100	0	2	8
	BRUCE B 5-8	0.00	100	0	2	8
	CERNAVODA	0.00	100	0	3	8
	DARL. A 1-2	200.00	100	0	2	8
	DOUGLAS PT.	0.00	100	0	2	8
	EMBALSE	300.00	100	***	3	8
	GENTILLY 2	200.00	100	***	4	4
	KNU 3	0.00	71	0	8	8
	PICK-A 1-4	0.00	100	0	2	8
	PICK-B 5-8	0.00	100	0	2	8
	POINT LEPREAU	200.00	100	0	2	8
LWGR	BELOYARSK 1	*******	***	***	***	***
	BELOYARSK 2	*******	***	***	***	***
	CHERNOBYL 1-2	*******	***	***	***	***
	CHERNOBYL 3-4	*******	***	***	***	***
	IGNALINA 1	*******	***	***	***	***
	IGNALINA 2	*******	***	***	***	***
	KOSTROMA 1	*******	***	***	***	***
	KURSK 1-2	*******	***	***	***	***
	KURSK 3	*******	***	***	***	***
	KURSK 4	*******	***	***	***	***
	LENINGRAD 1-2	*******	***	***	***	***
	LENINGRAD 3-4	*******	***	***	***	***
	SMOLENSK 1-2	*******	***	***	***	***
	TROITSK	*******	***	***	***	***
PWR	AKTASH 1	*******	***	***	***	***

A2. SPEED CONTROL CHARACTERISTICS BY SENSITIVITY AND DEADBAND - 2 MAY 1985

'***' and '?' = Data not provided in questionnaire reply, (*** = numerical, ? = text)

REACTOR TYPE	STATION	SPEED CONTROL DEADBAND +-mHZ	NORMAL VALVE LIMIT (%)	REGULATING BAND %MCR	DROOP MIN (% MCR)	DROOP MAX (% MCR)
	AKTASH 2	*******	***	***	***	***
	ALMARAZ 1	500.00	100	10	5	10
	ALMARAZ 2	500.00	100	10	5	10
	ANGRA D REIS 1	130.00	100	5	4	4
	ANGRA D REIS 2,3	130.00	100	5	4	4
	ARKANSAS N1-1	0.00	100	0	5	5
	ARKANSAS N1-2	0.00	100	5	5	5
	ARMYANSK 1-2	*******	***	***	5	5
	ASCO 1-2	333.00	95	0	1	10
	BEAVER VALLEY 1	0.00	100	9	3	3
	BEAVER VALLEY 2	0.00	100	9	3	3
	BELLEVILLE 1-2	0.00	100	7	4	4
	BEZNAU 1	200.00	100	0	2	10
	BEZNAU 2	200.00	100	0	2	10
	BIBLIS A	0.00	100	0	2	8
	BIBLIS B	0.00	100	0	2	8
	BORSSELE	120.00	100	0	4	4
	BROKDORF	*******	100	***	2	8
	BUGEY 2-3	0.00	100	7	4	4
	BUGEY 4-5	0.00	100	7	4	4
	CALLAWAY 1	0.00	100	5	5	5
	CATTENOM 1-2	0.00	100	7	4	4
	CATTENOM 3-4	0.00	100	7	4	4
	CHINON B1-2	0.00	100	7	4	4
	CHINON B3-4	0.00	100	7	4	4
	CHOOZ (SENA)	0.00	100	0	4	4
	CHOOZ B1	0.00	100	7	4	4
	CRUAS 1-4	0.00	100	7	4	4
	CRYSTAL R.3	300.00	100	5	3	7
	DAMPIERRE 1-4	0.00	100	7	4	4
	DAVIS-BESSE 1	0.00	100	1	5	5
	DOEL 1-2	0.00	100	0	4	4
	DOEL-3	0.00	100	0	4	4
	DOEL-4	0.00	100	0	4	4
	DONALD C COOK 1	0.00	100	0	2	7
	DONALD C COOK 2	0.00	100	0	2	7
	EMSLAND	*******	100	***	2	8
	FESSENHEIM 1-2	0.00	100	7	4	4
	FLAMANVILLE 1-2	0.00	100	7	4	4
	GENKAI 1	0.00	100	0	5	5
	GENKAI 2	0.00	100	0	5	5

A2. SPEED CONTROL CHARACTERISTICS BY SENSITIVITY AND DEADBAND - 2 MAY 1985

'***' and '?' = Data not provided in questionnaire reply. (*** = numerical, ? = text)

REACTOR TYPE	STATION	SPEED CONTROL DEADBAND +-mHZ	NORMAL VALVE LIMIT (%)	REGULATING BAND %MCR	DROOP MIN (% MCR)	DROOP MAX (% MCR)
	GKN 1	50.00	100	0	2	8
	GKN 2	*******	100	***	2	8
	GOESGEN	0.00	100	0	5	5
	GOLFECH 1	0.00	100	7	4	4
	GRAFENRHEINFELD	50.00	100	0	2	8
	GRAVELINES B1-4	0.00	100	7	4	4
	GRAVELINES C5-6	0.00	100	7	4	4
	GROHNDE	*******	100	***	2	8
	IKATA 1	410.00	100	0	5	5
	IKATA 2	410.00	100	0	5	5
	IKATA 3	*******	100	0	***	***
	ISAR 2	*******	100	***	2	8
	JOSE CABRERA	*******	100	***	***	***
	KALININ 1-2	*******	***	***	***	***
	KALININ 3-4	*******	***	***	***	***
	KHMELNITSKI 1	*******	***	***	***	***
	KHMELNITSKI 2	*******	***	***	***	***
	KNU 1	0.00	60	0	2	119
	KNU 2	100.00	57	0	4	68
	KNU 5	0.00	100	0	3	100
	KNU 6	0.00	100	***	3	100
	KNU 7	200.00	100	0	***	***
	KNU 8	200.00	100	0	3	100
	KNU 9-10	0.00	100	0	2	17
	KOEBERG 1-2	0.00	65	10	2	10
	KOLA 1-2	*******	***	***	5	5
	KOLA 3-4	*******	***	***	5	5
	LE BLAYAIS 1-4	0.00	100	7	4	4
	LOVIISA 1,2	100.00	95	7	1	5
	MAANSHAN 1	0.00	100	0	1	10
	MAANSHAN 2	0.00	100	0	1	10
	MIHAMA 1	0.00	100	0	5	5
	MIHAMA 2	0.00	100	0	5	5
	MIHAMA 3	0.00	100	0	5	5
	MUELHEIM-K	0.00	100	0	5	5
	NEFTYKAMSK 1	*******	***	***	***	***
	NIKOLAYEV 1	*******	***	***	***	***
	NIKOLAYEV 2	*******	***	***	***	***
	NIKOLAYEV 3-4	*******	***	***	***	***
	NIZHNEKAMSK 1-2	*******	***	***	***	***
	NIZHNEKAMSK 3-4	*******	***	***	***	***

A2. SPEED CONTROL CHARACTERISTICS BY SENSITIVITY AND DEADBAND - 2 MAY 1985

'***' and '?' = Data not provided in questionnaire reply, (*** = numerical, ? = text)

REACTOR TYPE	STATION	SPEED CONTROL DEADBAND +-mHZ	NORMAL VALVE LIMIT (%)	REGULATING BAND %MCR	DROOP MIN (% MCR)	DROOP MAX (% MCR)
	NOGENT 1-2	0.00	100	7	4	4
	NOVOVORONEZH 1	*******	***	***	***	***
	NOVOVORONEZH 2	*******	***	***	***	***
	NOVOVORONEZH 3-4	*******	***	***	5	5
	NOVOVORONEZH 5	*******	***	***	***	***
	OBRIGHEIM	0.00	100	0	2	8
	ODESSA 1-2	*******	***	***	***	***
	OHI 1-2	0.00	100	0	5	5
	OLT	0.00	100	0	5	5
	PAKS 1-2	200.00	88	3	5	5
	PAKS 3-4	200.00	88	3	5	5
	PALUEL 1-2	0.00	100	7	4	4
	PALUEL 3-4	0.00	100	7	4	4
	PENLY 1-2	0.00	100	7	4	4
	PHILIPPSBURG 2	0.00	100	***	2	8
	RINGHALS 2	0.00	100	0	4	4
	RINGHALS 3-4	0.00	100	0	4	4
	ROVNO 1-2	*******	***	***	5	5
	ROVNO 3	*******	***	***	***	***
	S ALBAN 1-2	0.00	100	7	4	4
	S LAURENT B1-2	0.00	100	7	4	4
	SALEM 1	300.00	85	10	4	4
	SALEM 2	300.00	85	10	4	4
	SENDAI 1	0.00	100	0	5	5
	SENDAI 2	0.00	100	0	5	5
	SEQUOYAH 1-2	0.00	100	0	5	5
	ST. LUCIE 1-2	66.00	100	0	5	5
	STADE	200.00	100	0	2	8
	TAKAHAMA 1	0.00	100	0	5	5
	TAKAHAMA 2	0.00	100	0	5	5
	TAKAHAMA 3-4	0.00	100	0	5	5
	TATAR 1	*******	***	***	***	***
	TIHANGE-1	0.00	100	0	4	4
	TIHANGE-2	0.00	100	0	4	4
	TIHANGE-3	0.00	100	0	4	4
	TMI-1	0.00	100	0	5	5
	TOMARI 1	0.00	85	5	3	5
	TOMARI 2	0.00	85	0	3	5
	TRICASTIN 1-4	0.00	100	7	4	4
	TRILLO 1	*******	100	***	***	***
	TRINO VERCL.	*******	100	0	4	4

A2. SPEED CONTROL CHARACTERISTICS BY SENSITIVITY AND DEADBAND - 2 MAY 1985

'###' and '?' = Data not provided in questionnaire reply. (### = numerical, ? = text)

REACTOR TYPE	STATION	SPEED CONTROL DEADBAND +-mHZ	NORMAL VALVE LIMIT (%)	REGULATING BAND %MCR	DROOP MIN (% MCR)	DROOP MAX (% MCR)
	TROJAN	0.00	100	0	3	8
	TSIMLYANSK 1	#######	###	###	###	###
	TSIMLYANSK 2	#######	###	###	###	###
	TSURUGA 2	0.00	100	0	5	5
	TURKEY PT 3-4	0.00	100	0	5	5
	UNTERWESER	0.00	100	0	2	8
	VANDELLOS 2	#######	100	0	###	###
	WATERFORD SES 3	67.00	100	0	1	10
	WOLF CREEK	0.00	100	5	5	5
	ZAPOROZHE 1	#######	###	###	###	###
	ZAPOROZHE 2-4	#######	###	###	###	###
	ZION 1-2	0.00	100	1	3	7

A3. LOAD CONTROL - RANGE AND FLEXIBILITY - ? MAY 1985

'***' and '?' = Data not provided in questionnaire reply. (*** = numerical, ? = text)

REACTOR TYPE	STATION	YEAR COMMISSIONED	SECONDARY LOAD CONTROL	UTILISATION NOW	UTILISATION FUTURE	CONTROL RANGE MIN (%MCR)	MAX	AV. LOADING RATE %MCR/MIN
BWR	ALTO LAZIO 1-2	1990	YES TFR	NO	?	35	100	10.00
	BARSEBECK 1-2	1977	YES TFR	NO	NO	60	100	1.00
	BROWNS FERRY 1	1977	NO TFR	NO	NO	10	100	1.00
	BRUNSBUETTEL	1976	YES TFR/CC	NO	?	10	100	15.00
	CAORSO	1981	YES TFR	NO	NO	***	100	******
	CHINSHAN 1	1978	YES TFR	YES	YES	65	100	3.00
	CHINSHAN 2	1979	YES TFR	YES	YES	65	100	3.00
	CLINTON	1986	YES ?	NO	NO	50	100	10.00
	COFRENTES	1984	YES RFT	NO	YES	50	100	50.00
	DODEWAARD	1968	YES TFR	NO	NO	50	100	10.00
	DRESDEN 2-3	1971	YES TFR	NO	YES	70	100	1.00
	ENRICO FERMI 2	1985	YES TFR	NO	NO	30	100	10.00
	F. DAIICHI 1	1971	YES TFR	NO	NO	30	100	10.00
	F. DAIICHI 2	1974	YES TFR	NO	NO	30	100	10.00
	F. DAIICHI 3	1976	YES TFR	NO	NO	30	100	10.00
	F. DAIICHI 4-5	1978	YES TFR	NO	NO	30	100	10.00
	F. DAIICHI 6	1979	YES TFR	NO	NO	30	100	10.00
	F. DAINI 1	1982	YES TFR	NO	NO	30	100	10.00
	F. DAINI 2	1984	YES TFR	NO	NO	30	100	10.00
	F. DAINI 3	1985	YES TFR	NO	NO	30	100	10.00
	F. DAINI 4	1987	YES TFR	NO	NO	30	100	10.00
	FORSMARK 1-2	1981	YES TFR	NO	NO	75	100	1.50
	FORSMARK 3	1985	YES TFR	NO	NO	70	100	1.50
	GUNDREM. 2B 2C	1984	YES TF/CC	NO	NO	10	100	******
	HAMAOKA 1	1976	YES RFT	NO	?	30	100	10.00
	HAMAOKA 2	1978	YES RFT	NO	NO	30	100	10.00
	HAMAOKA 3	1987	YES RFT	NO	NO	30	100	10.00
	ISAR 1	1977	YES TFR/CC	NO	?	10	100	60.00
	KASHIWAZAKI-K1	1985	YES TFR	NO	NO	30	100	10.00
	KASHIWAZAKI-K2-3	1990	YES TFR	NO	NO	30	100	10.00
	KRUEMMEL	1983	YES TF/CC	NO	?	10	100	******
	KUOSHENG 1	1981	YES TFR	YES	YES	60	100	15.00
	KUOSHENG 2	1983	YES TFR	YES	YES	60	100	15.00
	LA SALLE C 1	1982	YES TFR	NO	YES	70	100	1.00
	LA SALLE C 2	1984	YES TFR	NO	YES	70	100	1.00
	LEIBSTADT	1984	YES RFT	NO	NO	0	100	5.00
	MUHLEBERG	1971	NO	NO	NO	50	100	5.00
	NINE MILE POINT	1969	NO	NO	NO	60	100	******
	OLKILUOTO 1,2	1982	YES TFR	NO	NO	60	100	3.00
	ONAGAWA 1	1984	YES TFR	NO	YES	30	100	10.00
	OSKARSHAMN 1	1972	YES TFR	NO	NO	70	100	1.00

A3. LOAD CONTROL - RANGE AND FLEXIBILITY - 2 MAY 1985

'***' and '?' = Data not provided in questionnaire reply, (*** = numerical, ? = text)

REACTOR TYPE	STATION	YEAR COMMISSIONED	SECONDARY LOAD CONTROL	UTILISATION NOW	UTILISATION FUTURE	CONTROL RANGE MIN (%MCR)	MAX	AV. LOADING RATE %MCR/MIN
	OSKARSHAMN 2	1975	YES TFR	NO	NO	70	100	1.00
	OSKARSHAMN 3	1985	YES TFR	NO	NO	70	100	1.00
	OYSTER CREEK 1	1969	YES TFR	NO	NO	15	100	15.00
	PHILIPPSBURG 1	1979	YES TFR/CC	NO	NO	10	100	60.00
	QUAD CITIES 1-2	1972	YES TFR	NO	YES	70	100	0.50
	RINGHALS 1	1976	YES TFR	NO	NO	70	100	1.00
	SHIMANE 1	1974	YES RFT	NO	NO	66	100	10.00
	SHIMANE 2	1989	YES TFR	NO	NO	30	100	10.00
	SM. GARONA	1971	? ?	?	?	***	100	******
	SUSQUEHANNA SE	1984	YES TFR	NO	NO	50	100	1.00
	TOKAI 2	1978	YES RFT	NO	NO	30	100	10.00
	TSURUGA 1	1970	YES RFT	YES	NO	30	100	10.00
	WINFRITH	1968	YES TFR/RFT	NO	NO	30	100	2.00
	WUERGASSEN	1971	YES TFR/CC	NO	YES	10	100	60.00
FBR	BELOYARSK 3	1981	YES RFT/TFR	?	?	10	100	5.00
	CREYS-MALVILLE	1986	YES CC	NO	NO	10	100	10.00
	DOUNREAY PFR	1975	YES RFT	NO	NO	20	100	10.00
	KALKAR	1986	YES CC	NO	NO	30	100	10.00
	MARCOULE PHENIX	1973	NO	NO	NO	20	100	0.10
	MONJU	1990	YES TFR	NO	NO	40	100	5.00
	SHEVCHENKO	1973	YES RFT/TFR	?	?	10	100	5.00
GCR	BERKELEY	1962	NO	NO	NO	50	100	2.00
	BRADWELL	1962	NO	NO	NO	50	100	2.00
	BUGEY 1	1972	YES TFR	NO	NO	50	100	2.00
	CALDER HALL	1956	NO	NO	NO	50	100	2.00
	CHAPELCROSS	1960	NO	NO	NO	50	100	2.00
	† CHINON 2	1964	NO RFT	NO	NO	50	100	2.00
	CHINON 3	1966	YES RFT	NO	NO	50	100	2.00
	DUNGENESS A	1965	YES RFT	NO	NO	50	100	4.00
	DUNGENESS B	1983	YES TFR	NO	NO	20	100	3.00
	HARTLEPOOL	1983	YES TFR	NO	NO	40	100	5.00
	HEYSHAM 1	1983	YES TFR	NO	NO	40	100	5.00
	HEYSHAM 2	1988	YES TFR	NO	NO	30	100	5.00
	HINKLEY. PT A	1965	NO	NO	NO	50	100	2.00
	HINKLEY. PT B	1976	YES RFT	NO	NO	40	100	5.00
	HUNTERSTON A	1964	NO	NO	NO	50	100	2.00
	HUNTERSTON B	1976	YES RFT	NO	NO	40	100	5.00
	LATINA	1964	NO	NO	NO	50	100	2.00
	† MONTS D'ARREE	1966	NO	NO	NO	50	100	2.00

A3. LOAD CONTROL - RANGE AND FLEXIBILITY - 2 MAY 1985

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REACTOR TYPE	STATION	YEAR COMMISSIONED	SECONDARY LOAD CONTROL	UTILISATION NOW	UTILISATION FUTURE	CONTROL RANGE MIN (%MCR)	MAX	AV. LOADING RATE %MCR/MIN
	OLDBURY	1967	YES RFT	NO	NO	50	100	4.00
	S LAURENT A1-2	1971	? ?	NO	NO	50	100	2.00
	SIZEWELL	1966	YES TFR	NO	NO	50	100	4.00
	TOKAI 1	1966	YES RFT	NO	NO	50	100	2.00
	TORNESS	1988	YES TFR	NO	NO	30	100	5.00
	TRAWSFYNYDD	1965	YES TFR	NO	NO	50	100	2.00
	VANDELLOS 1	1972	? ?	?	?	***	100	******
	WYLFA	1971	YES TFR	NO	NO	50	100	5.00
HTGR	UENTROP	1984	YES CC	NO	NO	30	100	******
HWR	ATUCHA 1	1974	YES TFR/CC	NO	YES	15	100	10.00
	ATUCHA 2	1987	YES TFR/CC	NO	YES	20	100	5.00
	BRUCE A 1-4	1978	YES RFT	NO	YES	60	100	10.00
	BRUCE B 5-8	1987	YES RFT	NO	YES	60	100	10.00
	CERNAVODA	1990	YES RFT	NO	YES	10	100	10.00
	DARL. A 1-2	1988	YES RFT	NO	YES	60	100	10.00
	DOUGLAS PT.	1968	YES TFR	NO	NO	50	100	5.00
	EMBALSE	1983	YES RFT	YES	YES	60	100	10.00
	GENTILLY 2	1982	YES RFT	NO	YES	60	100	10.00
	KNU 3	1983	YES TFR	NO	NO	10	100	10.00
	PICK-A 1-4	1973	YES TFR	NO	YES	60	100	12.00
	PICK-B 5-8	1985	YES RFT	NO	YES	60	100	12.00
	POINT LEPREAU	1982	YES RFT	NO	YES	60	100	10.00
LWGR	BELOYARSK 1	1964	YES RFT/TFR	?	?	10	100	5.00
	BELOYARSK 2	1967	YES RFT/TFR	?	?	10	100	5.00
	CHERNOBYL 1-2	1978	YES RFT/TFR	?	?	10	100	5.00
	CHERNOBYL 3-4	1984	YES RFT/TFR	?	?	10	100	5.00
	IGNALINA 1	1984	YES RFT/TFR	?	?	10	100	5.00
	IGNALINA 2	1986	YES RFT/TFR	?	?	10	100	5.00
	KOSTROMA 1	1987	YES RFT/TFR	?	?	10	100	5.00
	KURSK 1-2	1979	YES RFT/TFR	?	?	10	100	5.00
	KURSK 3	1983	YES RFT/TFR	?	?	10	100	5.00
	KURSK 4	1986	YES RFT/TFR	?	?	10	100	5.00
	LENINGRAD 1-2	1975	YES RFT/TFR	?	?	10	100	5.00
	LENINGRAD 3-4	1981	YES RFT/TFR	?	?	10	100	5.00
	SMOLENSK 1-2	1984	YES RFT/TFR	?	?	10	100	5.00
	TROITSK	1958	YES RFT/TFR	?	?	10	100	5.00
PWR	AKTASH 1	1986	YES RFT/TFR	?	?	10	100	5.00

A3. LOAD CONTROL - RANGE AND FLEXIBILITY - ? MAY 1985

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REACTOR TYPE	STATION	YEAR COMMISSIONED	SECONDARY LOAD CONTROL	UTILISATION NOW	UTILISATION FUTURE	CONTROL RANGE MIN (%MCR)	MAX	AV. LOADING RATE %MCR/MIN
	AKTASH 2	1988	YES RFT/TFR	?	?	10	100	5.00
	ALMARAZ 1	1980	YES RFT	YES	YES	15	100	5.00
	ALMARAZ 2	1983	YES RFT	YES	YES	15	100	5.00
	ANGRA D REIS 1	1982	YES RFT	NO	NO	15	100	5.00
	ANGRA D REIS 2,3	1990	YES RFT	NO	NO	15	100	5.00
	ARKANSAS N1-1	1974	YES TFR	NO	NO	15	100	5.00
	ARKANSAS N1-2	1980	YES TFR	NO	NO	15	100	5.00
	ARMYANSK 1-2	1980	YES RFT/TFR	?	?	10	100	5.00
	ASCO 1-2	1985	YES RFT	NO	YES	15	100	0.50
	BEAVER VALLEY 1	1976	YES RFT	NO	NO	20	100	******
	BEAVER VALLEY 2	1986	YES RFT	NO	NO	20	100	******
	BELLEVILLE 1-2	1987	YES RFT	NO	YES	20	100	5.00
	BEZNAU 1	1969	YES RFT	NO	NO	15	100	5.00
	BEZNAU 2	1971	YES RFT	NO	NO	15	100	5.00
	BIBLIS A	1974	YES CC	NO	NO	10	100	15.00
	BIBLIS B	1976	YES CC	NO	NO	10	100	10.00
	BORSSELE	1973	YES CC	NO	NO	30	100	10.00
	BROKDORF	1986	YES CC	NO	?	20	100	0.00
	BUGEY 2-3	1978	YES RFT	NO	YES	20	100	5.00
	BUGEY 4-5	1979	YES RFT	NO	YES	20	100	2.00
	CALLAWAY 1	1984	YES CC	NO	NO	15	100	10.00
	CATTENOM 1-2	1986	YES RFT	NO	YES	20	100	5.00
	CATTENOM 3-4	1989	YES RFT	NO	YES	20	100	5.00
	CHINON B1-2	1982	YES RFT	NO	YES	20	100	5.00
	CHINON B3-4	1987	YES RFT	NO	YES	20	100	5.00
	CHOOZ (SENA)	1967	NO RFT	NO	NO	50	100	2.00
	CHOOZ B1	1990	YES RFT	NO	YES	20	100	5.00
	CRUAS 1-4	1984	YES RFT	NO	YES	20	100	5.00
	CRYSTAL R.3	1977	YES CC	NO	NO	15	100	3.00
	DAMPIERRE 1-4	1982	YES RFT	NO	YES	20	100	5.00
	DAVIS-BESSE 1	1977	NO	NO	NO	***	100	5.00
	DOEL 1-2	1975	YES RFT	NO	?	15	100	5.00
	DOEL-3	1982	YES RFT	NO	?	15	100	5.00
	DOEL-4	1985	YES RFT	?	?	15	100	5.00
	DONALD C COOK 1	1975	NO	NO	NO	***	100	5.00
	DONALD C COOK 2	1978	NO	NO	NO	***	100	5.00
	EMSLAND	1988	YES CC	NO	YES	20	100	******
	FESSENHEIM 1-2	1978	YES RFT	NO	YES	20	100	5.00
	FLAMANVILLE 1-2	1985	YES RFT	NO	YES	20	100	5.00
	GENKAI 1	1975	YES RFT	NO	?	15	100	5.00
	GENKAI 2	1981	YES RFT	NO	?	15	100	5.00

A3. LOAD CONTROL - RANGE AND FLEXIBILITY - 2 MAY 1985

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REACTOR TYPE	STATION	YEAR COMMISSIONED	SECONDARY LOAD CONTROL	UTILISATION NOW	UTILISATION FUTURE	CONTROL RANGE (%MCR) MIN	MAX	AV. LOADING RATE %MCR/MIN
	GKN 1	1976	YES CC	NO	YES	40	100	10.00
	GKN 2	1989	YES CC	NO	?	20	100	######
	GOESGEN	1979	YES CC	NO	NO	20	100	5.00
	GOLFECH 1	1990	YES RFT	NO	YES	20	100	5.00
	GRAFENRHEINFELD	1981	YES RFT/CC	NO	YES	30	100	10.00
	GRAVELINES B1-4	1981	YES RFT	NO	YES	20	100	5.00
	GRAVELINES C5-6	1985	YES RFT	NO	YES	20	100	5.00
	GROHNDE	1984	YES CC	NO	?	20	100	######
	IKATA 1	1977	YES RFT	NO	YES	15	100	5.00
	IKATA 2	1982	YES RFT	NO	YES	15	100	5.00
	IKATA 3	1990	YES RFT	NO	YES	15	100	5.00
	ISAR 2	1988	YES CC	NO	?	20	100	######
	JOSE CABRERA	1969	? ?	?	?	###	100	######
	KALININ 1-2	1984	YES RFT/TFR	?	?	10	100	5.00
	KALININ 3-4	1990	YES RFT/TFR	?	?	10	100	5.00
	KHMELNITSKI 1	1984	YES RFT/TFR	?	?	10	100	5.00
	KHMELNITSKI 2	1986	YES RFT/TFR	?	?	10	100	5.00
	KNU 1	1978	YES RFT	NO	NO	20	100	5.00
	KNU 2	1983	YES RFT	NO	YES	15	100	5.00
	KNU 5	1985	YES RFT	NO	YES	15	100	3.00
	KNU 6	1985	YES RFT	NO	NO	15	100	5.00
	KNU 7	1986	NO RFT	NO	NO	15	100	5.00
	KNU 8	1987	YES RFT	NO	NO	15	100	5.00
	KNU 9-10	1989	YES RFT	NO	NO	15	100	5.00
	KOEBERG 1-2	1984	YES RFT	YES	YES	15	100	5.00
	KOLA 1-2	1974	YES RFT/TFR	?	?	10	100	5.00
	KOLA 3-4	1984	YES RFT/TFR	?	?	10	100	5.00
	LE BLAYAIS 1-4	1983	YES RFT	NO	YES	20	100	5.00
	LOVIISA 1,2	1980	YES RFT	YES	YES	15	100	1.00
	MAANSHAN 1	1984	YES RFT	YES	YES	50	100	5.00
	MAANSHAN 2	1985	YES RFT	NO	YES	50	100	5.00
	MIHAMA 1	1970	YES RFT	NO	YES	15	100	5.00
	MIHAMA 2	1972	YES RFT	NO	YES	15	100	5.00
	MIHAMA 3	1976	YES RFT	NO	YES	15	100	5.00
	MUELHEIM-K	1985	YES CC	NO	?	15	100	10.00
	NEFTYKAMSK 1	1989	YES RFT/TFR	?	?	10	100	5.00
	NIKOLAYEV 1	1984	YES RFT/TFR	?	?	10	100	5.00
	NIKOLAYEV 2	1987	YES RFT/TFR	?	?	10	100	5.00
	NIKOLAYEV 3-4	1990	YES RFT/TFR	?	?	10	100	5.00
	NIZHNEKAMSK 1-2	1986	YES RFT/TFR	?	?	10	100	5.00
	NIZHNEKAMSK 3-4	1988	YES RFT/TFR	?	?	10	100	5.00

A4. LOAD REJECTION AND TURBINE BYPASS CONTROL, ISOLATED OPERATION AND PROVISION FOR TRIPPING TO HOUSE LOAD - 2 MAY 1985

SS = STARTUP / SHUTDOWN. LR = LOAD REJECTION. '***' and '?' = data not provided in questionnaire reply (*** = numeric , ? = text)

REACTOR TYPE	STATION	BYPASS CAPACITY %MCR	BYPASS CONTROL SS	BYPASS CONTROL LR	ISOLATED OPERATION	PROVISION FOR TTH / RESIDUAL LOAD / AND TIME PERIOD
BWR	ALTO LAZIO 1-2	35	YES	YES	YES	YES/35%/UNLIMITED
	BARSEBECK 1-2	100	YES	YES	YES	YES/3.2%/UNLIMITED
	BROWNS FERRY 1	60	YES	YES	NO	NO
	BRUNSBUETTEL	90	YES	YES	YES	YES/5%/UNLIMITED
	CAORSO	32	YES	YES	YES	NORMALLY, NO, YES BELOW 30% MCR
	CHINSHAN 1	25	YES	YES	NO	NO
	CHINSHAN 2	25	YES	YES	NO	NO
	CLINTON	35	YES	YES	NO	NO
	COFRENTES	35	YES	YES	NO	NO
	DODEWAARD	100	YES	YES	YES	YES/4.5%/UNLIMITED
	DRESDEN 2-3	40	YES	YES	YES	NO
	ENRICO FERMI 2	27	YES	YES	NO	NO
	F. DAIICHI 1	105	YES	YES	YES	YES/5%/8
	F. DAIICHI 2	25	YES	YES	NO	NO
	F. DAIICHI 3	25	YES	YES	NO	NO
	F. DAIICHI 4-5	25	YES	YES	NO	NO
	F. DAIICHI 6	25	YES	YES	NO	NO
	F. DAINI 1	25	YES	YES	NO	NO
	F. DAINI 2	25	YES	YES	NO	NO
	F. DAINI 3	100	YES	YES	NO	YES/5%/8
	F. DAINI 4	100	YES	YES	NO	YES/5%/8
	FORSMARK 1-2	100	YES	YES	YES	YES/3%/UNLIMITED
	FORSMARK 3	100	YES	YES	YES	YES/2%/UNLIMITED
	GUNDREM. 2B 2C	60	YES	YES	YES	YES/5%/UNLIMITED
	HAMAOKA 1	25	YES	YES	NO	NO
	HAMAOKA 2	25	YES	YES	NO	NO
	HAMAOKA 3	25	YES	YES	NO	NO
	ISAR 1	60	YES	YES	YES	YES/5%/UNLIMITED
	KASHIWAZAKI-K1	25	YES	YES	NO	NO
	KASHIWAZAKI-K2-3	100	YES	YES	YES	YES/5%/8
	KRUEMMEL	80	YES	YES	YES	YES/5%/UNLIMITED
	KUOSHENG 1	35	YES	YES	NO	NO
	KUOSHENG 2	35	YES	YES	NO	NO
	LA SALLE C 1	25	YES	YES	YES	NO
	LA SALLE C 2	25	YES	YES	YES	NO
	LEIBSTADT	110	YES	YES	YES	YES/5%/UNLIMITED
	MUHLEBERG	110	YES	YES	YES	YES/5%/UNLIMITED
	NINE MILE POINT	42	YES	YES	NO	NO
	OLKILUOTO 1,2	100	YES	YES	YES	YES/2.5%/24
	ONAGAWA 1	25	YES	YES	NO	NO
	OSKARSHAMN 1	100	YES	YES	YES	YES/3.3%/30MIN

APPENDIX A.5

SC39 WG04 POWER PLANT CONTROL
SUB-GROUP : CONTROL OF NUCLEAR PLANT

MARCH 1983
(Revised Dec.1984)

re: QUESTIONNAIRE - Control of Nuclear Generation

NOTES OF GUIDANCE

INTRODUCTION

Working Group 04 of CIGRE# Study Committee 39 is undertaking a survey of present and planned future practices on the subject of power and frequency regulation by nuclear units. To this end we have prepared the following questionnaire which we hope will provide factual information on this subject of importance to utility system planning and operations. Your response to the questionnaire will be greatly appreciated.

To make the questions clear in the intended context, the following preamble is included which describes terminology and the type of control commmonly used on fossil-fired thermal generation. It is generally known from the manufacturer's literature that nuclear units can meet or excel response capabilities of normal fossil-fired units. However not well known is the actual use of these capabilities in various systems.

COUNTRY OF ORIGIN AND NSSS DATA   Qs (0) - (9)

Questions (0)-(6) relate to country of origin, Nuclear sites, year of commissioning, Station output-MWe, Generated/Export, number of reactors and type, and steam plant data - but, excluding manufacturers.

A description of the use of Bypass Valves in normal and grid-imposed emergency conditions is of interest. Questions (7)-(9) relate to the provisions for steam/water bypassing of the main turbines which are supplementary to the normal safety valves which are provided for boiler cooling under emergency conditions. Questions (8) and (9) distinguish between whether the Bypass provision is to cater for start-up and shutdown only, for emergency use (steam and load conservation) following load rejection, or for both.

SPEED AND LOAD CONTROL   Qs (10)-(16)

(A) Primary Control (10)-(14)

Every prime mover system, for safety and practical reasons, must have some type of speed control system in order to limit overspeed in case of load rejection and to allow satisfactory load sharing during parallel operation. The usual speed control acts therefore as a proportional control, opening and closing valves in response to speed error. Control action may be limited in the opening direction by the valve limit setting, but not in the closing direction due to obvious overspeed safety considerations.

Conference International des Grand Reseaux Electriques a Haute Tension
International Conference on large High Voltage Electric Systems.

Primary controls are sometimes deliberately desensitised through provision of a deadband on speed deviation so that valve motion does not occur for small speed error.

In the following questionnaire information is sought as to:

(i) Type of Governing Mode - whether HP only or HP + IP (Dual Mode) - Question (10) *

(ii) Whether primary control is with or without some deadband. If deadband is deliberately provided the extent of deadband for positive and negative speed errors in MHz is of interest - Question (11). If none, state 0. (See Figure attached).

(iii) Range of adjustment available for setting the overall Speed Droop in % of MCR - Question (12) : State minimum and maximum values.

(iv) The normal practice in regard to setting of the valve limit relative to the operating level, i.e. the positive margin in valve travel available to enable a response to frequency drops or to increases in power setter (load reference) adjustment - Question (13). If valves are normally fully open, state 100.

(v) Regulating Band, ± % MCR, normally exercised by primary control. Question (14). If plant is not intended to contribute, state 0.

(B) Secondary (automatic generation) control (15)-(16)

By secondary (automatic generation) control is meant the control action to change generation output by means other than primary control, i.e. by adjustments to the power (load) setting of the unit.

The generation change requirements of particular units is determined at the central dispatch centre and raise and lower commands are broadcast to the various units under control in the system, or alternatively implemented by manual control. Is it provided? - Question 15(a).

The methods of control implementation for NSSS's fall into three principal types - RFT, TFR or CC (defined below) : Question 15(b).

(i) Raise and lower pulses sent directly to the turbine power setting (speed/load changer). The resulting turbine valve positon changes cause changes in steam flow and hence MW output, and restoration of balance of energy input to match output plus storage changes follows through the steam generator controls. This philosophy is known as Reactor Following Turbine or RFT.

* This Question was deleted in the December '84 revision.

or (ii) Commands for changes in generation are routed to the steam generator control system and act first to change input energy to the boiler. The turbine valves, under pressure control, then move to match steam energy flow output to steam generation. This philosophy is called "Turbine Following Reactor" or TFR.

or (iii) A combination of the two above types of control wherein control action is routed simultaneously to reactor and turbine is labelled "Co-ordinated Controls" or CC.

FIGURE 7 illustrates the three basic philosophies.

Question (16) is aimed at identifying whether utilities operate the plant in a load-following manner (Secondary control) now - 16(a), and the intentions in the future - 16(b).

STATION RESPONSE CHARACTERISTICS Qs (17) - (19)

Important characteristics comprise:

(i) The range of power over which automatic control can be exercised (minimum and maximum values) and the loading rate - Question (17) in three parts.

(ii) The linearized response to a small step in demand. The general shape of this response should be drawn as function of time. Question (18) *

(iii) Load Limitations due to Xenon poisoning, fuel clad interaction, thermal stress, fuel cycle, etc. which prescribe the limits of daily load-following capability in terms of minimum and maximum load.

ISOLATED OPERATION (20)

In case the plant is left with some load but isolated from other sources of generation, can it continue in operation and if so, under what condition of load, and for how long? - Question (20).

TRIPPING TO HOUSE LOAD - TTH (21)

Is the plant kept in operation in the event of loss of electrical connections to the grid? If so, what is the minimum residual load and the time period? - Question (21).

In order to gather information on the above questions, it is requested that the attached questionnaire be filled out and returned to the indicated CIGRE Working Group member or as below # for corresponding members.

ADDITIONAL INFORMATION (22)

Any other comments?

For your guidance, sample answers to the questions are attached.

Sub-Group: Nuclear Plant Control

	QUESTIONS		ANSWERS
(0)	COUNTRY	(0)	
(1)	Name of Utility	(1)	
(2)	Name of Station	(2)	
(3)	Year Commissioned to High Power	(3)	
(4)a	Generated Power for Unit(s) - Total MW	(4)a	
(4)b	Exported Power for Unit(s) - Total MW	(4)b	
(5)a	Reactor Type	(5)a	
(5)b	Number of Reactors	(5)b	
(6)a	Number of Turbines Per Reactor.	(6)a	
(6)b	Is Steam Reheat Provided ?	(6)b	
(7)	Turbine Bypass to Condenser - Capacity in %MCR	(7)	
(8)	Is Steam Bypass to Condenser Provided for Start-Up and Shutdown ?	(8)	
(9)	Is Steam Bypass to Condenser Operated in Load Rejection Situations ?	(9)	
(11)	What is the Setting, if any, of a Primary Speed Control Deadband +-mHz ?	(11)	
(12)a	Turbine speed governor Droop Setting - Minimum Value (%)	(12)a	
(12)b	Turbine speed governor Droop Setting - Maximum Value (%)	(12)b	
(13)	Turbine Throttle Valve Position in Normal Operation (%)	(13)	
(14)	Load Regulation Band Above and Below Operating load, +-%MCR	(14)	
(15)a	Load control (Secondary) : Is it Provided ?	(15)a	
(15)b	Is Philosophy TFR, RFT, or CC ?	(15)b	
(16)a	Load Control (Secondary) - Is it used now ?	(16)a	
(16)b	Load Control (Secondary) - Is it to be used in the Future ?	(16)b	
(17)a	Auto-Control or Control Range : Minimum Value %MCR	(17)a	
(17)b	Auto-Control or Control Range : Maximum Value %MCR	(17)b	
(17)c	Flexibility : Average Loading Rate %MCR per Minute	(17)c	
(19)	Daily Load Following Capability - Minimum Value and Maximum Values %MCR	(19)	
(20)	Grid-Isolated Operation : Is it Possible to operate the Plant without a Grid Connection ?	(20)	
(21)a	Is Tripping to Houseload Provided ?	(21)a	
(21)b	Value of Residual Generated Load %MCR	(21)b	
(21)c	For how Long - Hours	(21)c	

(22) Any Other Comments ?

Note: "*****" and "?" = Data not provided in reply to questionnaire, (***** = numerical, ? = text)

Abbreviations: MW = Megawatts MCR = Maximum Continuous Rating TFR = Turbine Following Reactor
RFT = Reactor Following Turbine CC = Coordinated Control 777.0 = Unlimited

	QUESTIONS		ANSWERS
(0)	COUNTRY	(0)	GB
(1)	Name of Utility	(1)	CEGB
(2)	Name of Station	(2)	DUNGENESS B
(3)	Year Commissioned to High Power	(3)	1983-5
(4)a	Generated Power for Unit(s) - Total MW	(4)a	1320
(4)b	Exported Power for Unit(s) - Total MW	(4)b	1200
(5)a	Reactor Type	(5)a	AGR
(5)b	Number of Reactors	(5)b	2
(6)a	Number of Turbines Per Reactor.	(6)a	1
(6)b	Is Steam Reheat Provided ?	(6)b	YES
(7)	Turbine Bypass to Condenser - Capacity in %MCR	(7)	40
(8)	Is Steam Bypass to Condenser Provided for Start-Up and Shutdown ?	(8)	YES
(9)	Is Steam Bypass to Condenser Operated in Load Rejection Situations ?	(9)	YES
(11)	What is the Setting, if any, of a Primary Speed Control Deadband +-mHz ?	(11)	0
(12)a	Turbine speed governor Droop Setting - Minimum Value (%)	(12)a	1.0
(12)b	Turbine speed governor Droop Setting - Maximum Value (%)	(12)b	4.0
(13)	Turbine Throttle Valve Position in Normal Operation (%)	(13)	100
(14)	Load Regulation Band Above and Below Operating load, +-%MCR	(14)	0
(15)a	Load control (Secondary) : Is it Provided ?	(15)a	YES
(15)b	Is Philosophy TFR, RFT, or CC ?	(15)b	TFR
(16)a	Load Control (Secondary) - Is it used now ?	(16)a	NO
(16)b	Load Control (Secondary) - Is it to be used in the Future ?	(16)b	NO
(17)a	Auto-Control or Control Range : Minimum Value %MCR	(17)a	20
(17)b	Auto-Control or Control Range : Maximum Value %MCR	(17)b	100
(17)c	Flexibility : Average Loading Rate %MCR per Minute	(17)c	3.00
(19)	Daily Load Following Capability - Minimum Value and Maximum Values %MCR	(19)	20-100
(20)	Grid-Isolated Operation : Is it Possible to operate the Plant without a Grid Connection ?	(20)	YES
(21)a	Is Tripping to Houseload Provided ?	(21)a	YES
(21)b	Value of Residual Generated Load %MCR	(21)b	5
(21)c	For how Long - Hours	(21)c	1.0

(22) Any Other Comments ? GENERAL RE. QS 15&16 AND LOAD/FREQ. CONTROL: AGR NO REGULATION ENVISAGED FOR UK NUCLEAR PLANT UNTIL THE YEAR 2000+

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