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Tokamak Instrumentation and Controls

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FUSION ENERGY DIVISION

TOKAMAK INSTRUMENTATION AND CONTROLS

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ABSTRACT

Controls and instrumentation problems projected for the higher powered and longer pulse time tokamaks — problems accentuated by the increasing R&D thrusts toward higher beta and actively shaped plasmas — need increased emphasis. These projected controls problems are considered (along with integration with ongoing experiments, the hardware-experienced research teams, and the plasma theory and engineering personnel) and the higher priorities needs selected for early emphasis.

The three areas of study emphasis to date are:

1. Physics implications for controls,
2. Computer simulation, and
3. Shutdown/aborts.

This document reports on the FY 78 efforts (the first year of these studies) to address these problems. Transient scenario options for the startup of a tokamak are developed, and the implications for the control system are discussed. This document also presents a hybrid computer simulation (analog and digital) of the Impurity Study Experiment (ISX-B) which is now being used for corroborative controls investigations. The simulation will be expanded to represent a TNS/ETF machine. The scoping of the abort potential problem has been initiated, and the preliminary results which are presented show the need for increased emphasis in this area.

Additional discussions are presented on areas of study that were relegated to secondary attention in the early work because of the limited effort that could be applied. They include:

1. I&C systems description,
2. R&D needs assessment, and
3. Development "road maps" to meet TNS/ETF/DEMO needs.

Needs and plans for continuing studies are discussed, with emphasis on the near-term or FY 79 work that is starting. A particular device and schedule must be selected to be used as a planning target. The appropriate device used is The Next Step (TNS) or Engineering Test Facility (ETF) machine.

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The development of the information in this report has resulted from the first year's study by the authors, who benefited considerably from and were supported by a broad spectrum of technical expertise in the ORNL Fusion Program and in the ORNL Instrumentation and Controls Division, expertise that was readily available and cooperatively provided. Early encouragement by M. Roberts and continuing support by H. H. Haselton and D. Steiner are much appreciated. W. M. Stacey, Jr. provided valuable consultation. T. C. Jernigan and J. L. Dunlap contributed useful guidance from the experimenter's point of view. Finally, J. K. Ballou and T. J. Huxford provided very timely calculational assistance.

1. INTRODUCTORY SUMMARY

To date the control approaches in fusion have been largely pre-programmed and open-loop, but they have sufficed for the low power, short pulse experiments. The successful extrapolation of such approaches to higher power, long burning time machines appears improbable where serious constraints such as the avoidance of high energy wall loading must be satisfied throughout the operation. The need for active, closed-loop control systems has been underscored further by the emphasis on the high beta and plasma shapes that require active establishment and maintenance, the needs for burn control and refueling, and the possible need for feedback control of the plasma heating energy.

The instrumentation systems used on magnetic confinement fusion experiments include many innovative approaches that have successfully provided the information required from the tests. For the most part, these diagnostics are laboratory-type equipment which may require extensive modification if used in a reactor environment of The Next Step (TNS) or Engineering Test Facility (ETF) device. The data handling approaches to be used on experiments such as the Tokamak Fusion Test Reactor (TFTR) will provide an excellent base for supporting the definition of instrumentation and diagnostic needs for TNS/ETF, although the long pulse times and the ignition operation may place significant demands on the equipment's real-time and archival capabilities. It is expected that TNS/ETF will impose the most stringent requirements on the instrumentation and control equipment, and it is toward this device that these studies are focused.

The emphasis of prime importance in this study program is the definition of the physics transients needed in a complete burn cycle. Thus, the largest portion of the investigation was so applied and is reported in Sect. 3. The impact of potential abnormal operations such as hard shutdown and aborts, as well as the avoidance and amelioration of such impacts, is deemed an area that needs strong attention and is reported in Sect. 5. The simulation of a tokamak and its control system is a valuable developmental tool for testing design concepts — especially when the experiment is being operated in the same facility

as the I&C study. The verification of the simulation that can be achieved with actual tokamak operation, as well as the potential for getting real transient data to test some particular control hypotheses, provides an excellent corroborative climate for developing the model and expanding it towards the TNS/ETF needs. A discussion of the initial simulation development accomplished to date is reported in Sect. 4.

The summary highlights of the FY 78 work that is building a valuable foundation for I&C R&D leading to meeting tokamak reactor needs include:

1. Pertinent physics has been coalesced for studies of control implications.
2. Transient characteristics in startup phases have been scoped with evolving codes.
3. Simplified Impurity Study Experiment (ISX-B) control equations have been run on analog and some of the stability concerns identified.
4. A full set of uniform torus ISX-B horizontal control equations has been programmed for analog machine.
5. Preliminary studies showed plasma disruptions as an area needing increased study emphasis.
6. Plasma disruption effects have been identified (e.g., temperature distribution, eddy currents, thermal stress, number of disruptions tolerable).
7. R&D needs assessment has been initiated.
8. FY 79 plans have been recast based on work to date.

2. METHOD OF ACCOMPLISHMENT

The activity in this investigation is guided by the program plan, which consists of the six tasks shown below. It is expected that, during following phases of the work, some change in emphasis may be required and some tasks may be added, subtracted, or redirected based on new findings.

Task I - Prepare (and maintain current) the characterization of instrumentation and controls needs of future magnetic confinement fusion devices from experimental devices to power reactors. Define startup to shutdown/abort scenarios.

Task II - Prepare (and maintain current) the description of the physics to be controlled, the desired bounds for the variables, the most suitable parameters to be sensed for control and data handling, and the candidate sensor techniques to be considered.

Task III - Initiate the preparation of a control system hybrid computer model that will provide the "test bed" for the control approaches being considered and provide the insight into the evolution of optimum algorithms and control law selection.

Task IV - Maintain an integration with current tokamak experiments and utilize the pertinent results. Define experimentation that can be accomplished on present or near-term devices (e.g., ISX) and that will provide support and guidance to the theory-devised control laws.

Task V - Prepare (and maintain current) a development "road map" depicting the I&C R&D needs, time-phased, showing interfaces with device programs and indicating need dates and potential costs.

Task VI - Maintain communications with all pertinent developers, receive data on needs, results, and approaches, and provide status on this program's progress to all interested. Plan and organize appropriate presentations, workshops, and technical society participation.

A matrix is used as an aid in understanding the overall picture of tokamak operation. The primary plasma systems are shown horizontally and the time phases of a cycle are shown vertically in the simple matrix

example shown in Fig. 1. The toroidal field (TF), poloidal field (PF), and heating and fueling systems have direct impact on plasma behavior; external systems such as cooling, tritium handling, and energy storage are examples of systems with very weak, if any, direct interaction with the plasma. The cycle is broken down into five phases — prebreakdown, breakdown, startup, burn, and shutdown — which may be further subdivided for more specific operation scenarios.

The particular I&C problem solving approach being used for each matrix intersection, e.g., what the PF system must do during breakdown, can be described by Fig. 2, which depicts the steps to be taken and the iterative answers to be determined: what physics is desired, how it can be sensed, what knobs are available to control the physics, and what transfer functions are necessary to close the loops in an appropriately stable manner. The definition of the physics desired, including the spatial and temporal variations, plus the resulting impact on the controls and instrumentation is the prime priority task.

There may be a wide variety of physical requirements for a given system during different phases of the cycle — along a vertical line in the matrix. If these requirements vary too widely, either by function or time scales, the system may be further divided into several components which have strong interaction. An example of this is the PF system, which must act as the primary of a transformer for inducing the toroidal current, provide the vertical field, and perhaps provide a poloidal divertor. These functions may be satisfied by a single set of coils, with each coil contributing to each of the functions, or by separate sets of coils for each function.

There may be interaction between the primary systems during a given phase of the cycle — along a horizontal line in the matrix. For instance, the pressure in the plasma increases when the plasma is heated, and the vertical field (VF) system must respond to maintain the plasma position.

The most benign system is the TF system, which should not require any active feedback; sensing systems and protective measures may, however, have to be incorporated for abnormal conditions. The most analyzed for control implications is the PF system. The framework for this analysis, as discussed in Sect. 4, has been applied to ohmic and beam heated

Fig. 1. Matrix of control issues.

Fig. 2. Control definition evolution.

plasmas. Extension to the burn phase of ignited plasmas should be relatively straightforward. The systems involving the most unknowns are the heating and fueling systems, which are discussed in Sect. 3.

The startup and shutdown phases place the greatest demand on the control system because of the dynamics and time scales involved. Controlled startup scenarios are discussed in Sect. 3. Abnormal shutdown or abort must be considered during all phases of the cycle because of the uncertainties in the physics. A greater understanding of the physics leading to major plasma disruptions has led to the analysis in Sect. 5.

The consideration of controls needs for the external systems has begun, and the initial findings are addressed briefly in Sect. 7.

3. PHYSICS IMPLICATIONS FOR CONTROLS

3.1 EVALUATION OF THE PHYSICS

Knowledge of the physical processes involved in tokamak plasma operation has expanded rapidly over the past several years. Models have become more sophisticated as confidence in the physics has increased, and new ideas have emerged for variations in the operating scenario. There is a continuing need for timely assessment of these new models and ideas in terms of their impact on, and the need for, control systems. The first device operating with a reacting plasma will still be experimental; it will not only have to handle the dynamics exhibited in currently operating ohmic and neutral beam heated devices, but it must also be designed to handle the anticipated behavior of plasmas with internal heating due to fusion. Only when very detailed knowledge of the physics over the entire operating range has been gathered can the control system approach optimization.

The emphasis in Sect. 3.2 is on evaluating the particle and energy balances in tokamaks, on determining whether density and temperature control is necessary, and on determining how this control may be accomplished. Recent theoretical and experimental results are discussed in terms of their impact on this evaluation. There are still many unknowns involved in assessing plasma behavior during burn. These unknowns and their potential range of influence on operating conditions and stability are outlined. Finally, the dynamics of controlled startup scenarios are discussed in Sect. 3.3.

3.2 HEATING AND FUELING CONTROL REQUIREMENTS

3.2.1 Thermal Stability Analysis

There are several levels of thermal equilibria for tokamak plasmas, with each one having very different stability criteria. The first is an equilibrium established between ohmic heating and all energy loss mechanisms. The equilibrium is very stable against temperature changes in the plasma because of the strong inverse temperature dependence of

the energy source, i.e., ohmic heating. If a temperature-independent energy source is added, such as with neutral beam injection, the temperature will rise to a new equilibrium. Two effects begin to change the nature of the equilibrium. First, the impact of ohmic heating on the stability decreases as ohmic heating becomes a smaller part of the total energy source. Because the power input becomes energy-independent, the temperature dependences of the loss mechanisms (radiation, conduction, and convection) begin to play a more prominent role in stability. If the conductive and convective energy losses decrease with increasing temperature as predicted by neoclassical theory, the temperature will rise to quite high values where increased radiative losses provide a stable equilibrium for the beam-driven system.

The addition of an internal source of energy as a result of the thermalization of fusion alphas in a deuterium-tritium (D-T) plasma dramatically changes the energy balance and, thus, the thermal stability. Above a temperature of 3 to 4 keV, the fusion rate rises rapidly. The 3.5-MeV alphas produced from fusion will thermalize in the plasma, if they are contained long enough, and provide an internal heat source. If the plasma is large enough, this internal source can exceed the radiative, conductive, and convective losses above a temperature called the ignition point. The plasma is inherently unstable to temperature perturbations at this point because of the rapid increase in the fusion rate with increasing temperature. The plasma will then undergo a thermal excursion to some higher temperature, called the burn point, where increased radiative losses (primarily from synchrotron radiation) and possibly enhanced conductive and convective losses overcome the increase in power from fusion. This equilibrium is inherently stable against temperature perturbations.

Although the ignition point is thermally unstable, it is attractive from a power production standpoint because of the relatively low temperature (8-15 keV), and high power density which can be attained in a beta limited system. For this reason, control at the ignition point has been discussed by many authors.¹⁻⁵ These analyses are typically global balances which evaluate the characteristic rates of excursion from equilibrium. The difficult part comes in proposing a viable control

mechanism. A particle source, an energy source, and compression-decompression have been suggested as potential candidates.

Adding cold fuel to the plasma has a cooling effect which reduces the fusion cross section. However, the total fusion rate increases as n^2 , which will generally offset the decrease in temperature and increase the rate of thermal excursion. For this scheme to work even in a global analysis, there must be a strong dependence on nonlinearities in the system. With a one-dimensional analysis which incorporates the radial density and temperature profiles, the most severe shortcomings are uncovered. The fusion energy source peaks strongly at the center of the plasma. To provide cooling, the fuel would have to be directly injected to the center of the plasma. A built-in delay time of the order of the particle confinement time is required to remove this excess fuel, which places a severe restriction on the capabilities of the control system. In addition, because the plasma profiles are not rigid, control about an unstable equilibrium at all radii would then require evaluation of spatial variations (profile shape control).

The use of an energy source has a more direct impact on the temperature. Control at the ignition point or control of a driven plasma^{6,7} with a source such as neutral beams should be fundamentally easier to accomplish than control with a particle source. The beam-driven plasma can be stable against temperature changes if operated in the Two-Component Torus (TCT) mode but requires relatively large amounts of power input.⁶ When applied to a plasma which is capable of igniting, the power requirements are reduced. However, as the fusion source from background D-T fusions begins to dominate, the plasma becomes thermally unstable, and a fast response time is required for the beam source to provide stability.⁷ The large amount of circulating power compared to net power output is perhaps the most limiting factor when this concept is applied to a fusion reactor. But the concept may be viable for exploring the ignition range in an experimental device while providing protection against thermal runaway.

A compression-decompression scenario has been recently proposed for control at the ignition point.⁸ Compression increases both density and temperature simultaneously and is reversible in the decompression stage,

thus allowing for fairly sensitive bulk control of the fusion power source. This concept appears attractive from the physics standpoint, but the impact on the design of the PF system and power consumption needs further analysis.

One-dimensional (radial) time-dependent simulations show that the plasma eventually reaches an equilibrium which is stable against temperature perturbations.^{9,10} Coupling between the particle and energy balances can, however, lead to a density related thermal instability when the diffusion coefficient is inversely proportional to density as in the trapped ion mode or anomalous scaling models.^{5,10} Feedback on the particle source leads to stabilization with a very simple feedback system. This is illustrated in Fig. 3 (taken from Ref. 10). Many uncertainties still exist in the physics which can dramatically change the plasma behavior at burn, as discussed in Sect. 3.2.5.

3.2.2. Density Profile Control

Control of the density and temperature profiles has been discussed in terms of control over the pressure profile and the corresponding beta profile. In both cases the profiles are determined by a balance between the sources and losses. Thus, mechanisms affecting the sources or losses would have an impact on the profiles. Although uncertainties exist in the transport models, some basic assessments can be made.

There are two primary candidates for fueling sources in fusion reactor systems: neutral gas puffing and pellet injection. Gas puffing gives very little control over the density profile because penetration in a large, dense plasma is poor. This results in a surface-type source with a large turnover in fuel (low fueling efficiency). However, it has the advantage of cooling the plasma edge, thus reducing impurity production, and it can be coupled to a feedback system with a fast response time. The gas source would likely have little impact on the density profile in the plasma interior, where balances between the diffusion processes will determine the profile. Pellet fueling, on the other hand, has the potential for deeper penetration and more control over the density profile.¹¹ However, oscillations in the density and temperature

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UWMAK-III

INSTANTANEOUS ALPHA THERMALIZATION

$$S_p(r,t) = S_p(t)(1-r/a)$$

— FEEDBACK ON

--- FEEDBACK OFF AT 4.5 sec

..... FEEDBACK OFF AT 5.4 sec

Fig. 3. Illustration of how feedback on the particle source can be used to stabilize a density related thermal instability in a power producing plasma.

profiles result from pellet fueling because of the discreteness of the pellets. These oscillations do not affect the pressure profile because the density and temperature oscillate out of phase; therefore, they should not adversely affect the plasma magnetohydrodynamic (MHD) stability.¹¹ Control of the pellet repetition rate would be required for control of the bulk density of the plasma. This is perhaps more difficult than control of the gas source. If so, some combination of gas puffing and pellets could be used for greater flexibility.

Control of the diffusion rate is another potential means of controlling the density profile. This control is more difficult to envision than particle source control because particle loss rates must be enhanced over nominal values to have any impact on the profile. A divertor zone at the edge of the plasma is an example of profile control by controlling losses. A low density region is established which may protect the wall from the plasma and prevent sputtered impurities from reaching the plasma.¹² Little control is provided over the density profile in the center of the plasma. In addition, high particle pumping rates are required for the divertor, particularly if gas puffing is used for the particle source.

3.2.3 Temperature Profile Control

Control of the temperature profile is conceptually the same as control of the density profile even though there are more internal sources and sinks in the energy balance: the sources or losses can be changed. The amount of energy conducted through a given flux surface, Q , at minor radius r is related to the temperature gradient and conduction coefficient κ as $Q = -\kappa(\partial T/\partial r)$.

Consider a perturbation with a localized energy source (e.g., microwaves) such that $Q' = Q + \Delta Q$. If a large part of the total energy loss is via conduction, Q already represents a considerable part of the total energy created in the plasma. To change the temperature gradient even by a few percent, ΔQ would have to be a nonnegligible part of the power plant's energy flow.

On the other hand, the conductivity κ can be locally enhanced, with the result that the temperature profile will be flattened locally. The temperature inside this region will also drop because the energy balance equation is parabolic. The introduction of ergodic zones by externally imposed helical magnetic perturbations¹³ or enhanced TF ripple¹⁴ may be used to increase κ . Ergodic zones are regions of finite radial extent in which the field lines are free to wander because of either internal perturbations or externally imposed perturbations, as suggested in Ref. 13. The effective radial transport coefficients (for both diffusion and conduction processes) become closer to the larger parallel transport coefficients along the magnetic field lines. However, there are limitations to the selectivity of this enhancement mechanism because of relationships between the local rotational transform and helicity of the external windings.¹³ The impact of the imposed perturbations on MHD stability must also be examined.

Poloidal field ripple effects have received a great amount of theoretical analysis (see literature cited in Ref. 14) but lack experimental verification. The theoretical analysis has the same basis as neoclassical theory and predicts enhanced ion conductivity near the edge of the plasma, where ripple is largest. The most collisionless ions, i.e., the high energy particles in the tail of the distribution, are preferentially lost. The use of ripple enhancement coils could provide experimental information, under controlled conditions, about the impact on the plasma edge conditions.

3.2.4 Recent Theoretical and Experimental Results

Nearly all advances in the understanding of tokamak physics have some direct or indirect influence on the assessment of control requirements for future systems. The ones considered here have all had significant impact already even though they have come along only in the past six months.

A theoretical understanding of major disruptions in tokamaks with predictions for the disruption time in larger devices¹⁵ has led to the analysis in Sect. 5. A direct impact is made on the assessment of the

ability to control a major disruption and the question of whether control is needed.

Two main areas of emphasis in ISX-A, pellet fueling and impurity flow reversal, have increased our knowledge of fueling and impurity control. The pellet fueling experiments substantially verified all theoretical predictions about fueling efficiency, pellet penetration, magnitude of density and temperature perturbations, and a voltage spike associated with injection of the pellet.¹⁶ The voltage spike is a result of the pellet's cooling the plasma, which increases the resistivity. The voltage is increased to maintain constant toroidal current. This is an example of the potential coupling between control systems; i.e., fueling with pellets introduces perturbations which the current control system (PF coils) must contend with. The impurity flow reversal experiments¹⁷ have demonstrated the potential for dynamic control of impurity flow. Neon impurity levels were reduced — in good agreement with theoretical predictions.

Experiments in the Princeton Large Torus (PLT) have shown that ion conductivity remains nearly neoclassical at very low collisionality (no large increase in losses because of trapped particle microinstabilities) with high power neutral beam injection.¹⁸ This helps provide confidence in the modeling of high temperature plasmas and in the prediction of plasma behavior. The density in PLT was held constant via feedback on the gas puffing sources,¹⁹ a fact which demonstrates the potential impact of even a relatively simple particle control system.

3.2.5 The Uncertainties in "Steady-State" Operation

There are many processes which may play a role in determining the "steady-state" or burn phase of a reacting tokamak plasma and which may, therefore, influence the plasma's thermal and MHD stability. These processes must be more thoroughly understood before control needs can be adequately assessed.

Much emphasis is traditionally placed on understanding how the transport coefficients scale at higher temperatures and densities. A strong increase in energy loss with increasing temperature is required

to provide an equilibrium which is stable against temperature perturbations, as discussed in Sect. 3.2.1. The anomalous electron conductivity found in all tokamaks still lacks a good theoretical understanding even though many theories have been advanced. The theory of microinstabilities and their resulting transport is being rewritten as more detailed numerical calculations of growth rates proceed. MHD-induced instabilities are now perceived as playing an extremely important role in transport processes through sawtooth and internal disruptions.²⁰

The influence of MHD instabilities at the beta limit of a plasma may be the overall dominating process in steady-state operation. There are two possible ways in which the plasma may react when the beta limit is reached. Fluctuations may increase in much the same manner as sawteeth and internal disruptions, which simply redistribute the particles, energy, and current within the plasma, giving automatic stabilization of conditions. If this happens, there may be no need to worry about thermal runaway of the plasma beyond a beta limit. On the other hand, if the beta limit results in a disruption of the entire plasma in a very short time, control procedures will have to be instituted to ensure that the beta limit is avoided.

Cyclotron radiation is expected to increase strongly with increasing plasma temperature. Very sophisticated two- or three-dimensional analysis is required to determine the net loss of energy from cyclotron radiation and from the transport of energy through nonlocal absorption in a nonuniform plasma.²¹ If the cyclotron radiation loss is strong enough, it can not only provide a stable thermal equilibrium at relatively low density but also transfer energy from the plasma to the chamber walls in a manner which should minimize wall damage.

The diffusion rates for deuterium and tritium may be different because of the different masses of the two species. Neoclassical effects predict that tritium may be screened from the plasma; i.e., deuterium would diffuse inward while forcing the more massive tritium ions out of the plasma center. The effects of anomalous diffusion processes could, however, totally change this picture. Experimental determination of relative diffusion in hydrogen and deuterium discharges is proceeding but is complicated by the effect of recycling from the limiter and wall,

which may be saturated with gas. In the event of such unequal diffusion processes, pellets of varying composition, velocity, and size could conceivably be used to control the plasma composition.

3.3 DYNAMIC CONSIDERATIONS FOR STARTUP

Control during the startup phase requires special consideration because there are no nominal steady-state parameters within which the control system must operate. Instead, the control parameters will be ramped in time from initial to final values, which can cover a wide parameter range. A brief summary of traditional startup in tokamaks will be presented. Controlled startup scenarios which are receiving more theoretical analysis because of their potential beneficial impact on physics and technology assessments will then be discussed.

The traditional startup scenario for tokamak experiments involves filling the chamber with gas and applying a large toroidal voltage spike through preprogrammed current ramps in the ohmic heating (OH) coils. This toroidal current spike serves two primary purposes: it breaks down the gas to produce an ionized plasma, and it induces a toroidal current in the plasma. The toroidal current aids in the plasma confinement through the rotational transform and provides ohmic heating to burn through the radiation barriers of hydrogen and other low-Z impurities. There appears to be no satisfactory way to control where the plasma breaks down. In addition, the breakdown process may be responsible for introducing impurities into the plasma through interaction with the limiter and walls. Extrapolation of this scheme to an ignition sized tokamak indicates that peak voltages of several hundred volts and peak powers in the ohmic system of a few gigawatts will be required.²²

A method which may potentially reduce the pulse technology requirements in tokamak startup has been proposed by Peng and Borowski.²³ It involves the use of a microwave power source to break down the plasma locally and provide some preheating. The ohmic coil system will then have the primary responsibility of providing the toroidal current. The division of the functional requirements between the two systems should

alleviate the pulse voltage and power requirements in the ohmic system. Experimental investigation of this concept is planned for ISX-B.

Classical calculations of the current rise phase in tokamaks indicate that a skin current should form at the plasma edge as a result of the long time scale for relaxation of the skin current (skin time) as compared to the current rise time. It has been observed experimentally that disruptive processes are responsible for redistributing this skin current in the plasma and creating an influx of impurities through interaction with the limiter and/or chamber walls.²⁴ A model has been developed for transport simulation of skin current redistribution which results from MHD disruptions similar to those which produce sawtooth oscillations.^{25,26} As the safety factor q passes through integer values at the plasma edge during the current rise, disruptions occur which reconnect magnetic surfaces. Comparison of the model with observed experimental parameters shows good agreement.²⁵

Expanding radius startup scenarios have been examined by several authors as means of reducing the skin current effect.^{27,28} This reduction should, in turn, reduce the influx of impurities during startup. The evolution of the current profile for such a scheme is shown in Fig. 4 for a TNS plasma. During the first 0.5 sec, the plasma radius is held fixed at half its full bore size while the current is ramped up to 1 MA. A modest skin current forms because no MHD effects were incorporated into the calculation. Redistribution of the skin current as a result of MHD internal disruptions will probably occur during this phase, but introduction of impurities should be reduced because of the small plasma size and distance from the chamber wall. During the next 1.5 sec, the plasma is expanded to full size while the toroidal current is simultaneously increased through control of the applied voltage. A uniform current density is laid in such that the safety factor q at the limiter remains constant. Because neutral beam heating is used during the expansion phase, the plasma temperature is fairly high and the resistivity low. The current profile remains "frozen" into the plasma.

Relatively long startup times for the beam heating phase reduce the beam power requirements, as shown in Fig. 5 for a typical TNS plasma.

Fig. 4. Evolution of the toroidal plasma current profile in an expanding radius startup scenario. During the first 0.5 sec, the current is increased from 50 kA to 1 MA while the minor radius is fixed at half its final value. A skin current develops. During the next 1.5 sec, the radius is increased and the current is ramped to maintain nearly constant $q(a)$.

Fig. 5. Impact of the beam heating time t_b on the minimum power required for ignition in a TNS plasma. The energy confinement time is about 1 sec. For times longer than $t_b \geq 4-5 \tau$, there is little savings in beam power.

An optimal time is several energy confinement times. These times, which are relatively long compared to present experimental time scales, greatly improve the prospect for control analysis. Folding in the above considerations, an "optimal" startup scenario such as that shown in Fig. 6 can be envisioned. This scenario could form the basis for further dynamic control analysis.

Fig. 6. I. Breakdown and ramp current to desired q in small plasma.
 II. Ramp size and current simultaneously to "ignition-size" plasma. Small amount of heating may be used to maintain plasma temperature and low electrical resistivity.
 III. Heat "ignition-size" plasma with neutral beams.
 IV. Ramp size and current to final size while relying on fusion burn for excess heating. In TNS the "burn-size" is very near the "ignition-size," and this phase may be eliminated.

4. COMPUTER SIMULATION

The hybrid modeling task is designed to provide useful insights into the behavior of a tokamak power reactor under operational conditions. Preparation of a design aid as specific as a computer program demands hard input in the form of system equations and parameter values. In order to provide such input, interaction has been initiated with the designers of the ISX-B experiment. This gives entry to a source of hard design data, provides help to the experimental program, and eventually can yield model verification through comparisons of computer runs with experimental data.

The first system modeled is the horizontal plasma position control, operating through inner and outer OH windings. (Vertical positioning involves an additional control variable, the radial field current, and will be introduced later as an extension of the horizontal control program.) The equations are taken from an ISX-B program paper by Charlton, Swain, and Neilson.²⁹ Assuming that vertical deviation is zero and neglecting the equation for radial field control, we can write four equations summing voltages around four current loops.

Plasma current, I_1 (I_1 is a negative current):

$$L_{11}\dot{I}_1 + L_{12}\dot{I}_2 + L_{13}\dot{I}_3 + L_{15}\dot{I}_5 + (L'_{11}I_1 - L'_{12}I_2 + L'_{13}I_3 - L'_{15}I_5)R_1 = 0 \quad (1)$$

Inner ohmic heating current, I_2 :

$$L_{12}\dot{I}_1 + L_{22}\dot{I}_2 + L_{23}\dot{I}_3 + L_{25}\dot{I}_5 - L'_{12}I_1R_1 + R_2I_2 = V_2 \quad (2)$$

Outer ohmic heating current, I_3 :

$$L_{13}\dot{I}_1 + L_{23}\dot{I}_2 + L_{33}\dot{I}_3 - L_{35}\dot{I}_5 + L'_{13}I_1R_1 + R_3I_3 = V_3 \quad (3)$$

Symmetrical shell (eddy) current, I_5 :

$$L_{15}\dot{I}_1 + L_{25}\dot{I}_2 - L_{35}\dot{I}_3 + L_{55}\dot{I}_5 - L'_{15}I_1R_1 + R_5I_5 = 0 \quad (4)$$

Figure 7 shows the relative orientation of these currents. Circuit values for the L's and R's were provided by D. Swain.³⁰

The vertical field has contributions from all of the currents:

$$B_v = \sum f_i(R_1) I_i, \quad (5)$$

where $i = 1, 2, 3, 5$.

The f_i 's are actually functions of both horizontal (R_1) and vertical (Z) displacement, but our first analysis limits itself to the horizontal. Figure 8 illustrates the fit of actual values to the relations assumed for f_i vs R . These latter are:

$$\begin{aligned} f_1 &= -(8.85 - 7.65 R) \quad \text{for } R \leq 0.85 \text{ m} \\ &= -(5.54 - 3.75 R) \quad \text{for } R \geq 0.85 \text{ m} , \end{aligned} \quad (6a)$$

$$\begin{aligned} f_2 &= -(107.16 - 94.74 R) \quad \text{for } R \leq 0.85 \text{ m} \\ &= -(62.08 - 41.67 R) \quad \text{for } R \geq 0.85 \text{ m} , \end{aligned} \quad (6b)$$

$$f_3 = 34.35 R , \quad (6c)$$

$$f_5 = -3.32 . \quad (6d)$$

Horizontal position is a function of plasma current β_p and the vertical field,

$$R_1 = - \frac{I_1(2.1 + \beta_p)}{K_1 B_v} . \quad (7)$$

Fig. 7. Orientation of plasma and control currents.

Fig. 8. Approximations to f_i functions used in analog model. Points = precise values; solid curves = analog approximations.

K_1 is a function of the plasma elongation and, for initial programming, is assumed to be 1.29. Negative values for I_1 produce positive R_1 's.

β_p is the poloidal pressure ratio, and we are given values of 1.2 to 4.9. In some computer runs, β will be stepped or ramped as a test of the control system.

In the analog simulation:

Equation (1) is solved for $\dot{I}_1$;

I_1 is integrated to yield I_1 .

Equation (2) is solved for $\dot{I}_2$;

I_2 is integrated to yield I_2 .

Equation (3) is solved for $\dot{I}_3$;

I_3 is integrated to yield I_3 .

Equation (4) is solved for $\dot{I}_1 R_1$;

$I_1 R_1$ is divided by I_1 to yield R_1 ,

R_1 is integrated to yield R_1 .

Equation (7) is solved for β_v .

Equations (6) are solved to yield f_1 , f_2 , and f_3 .

All of the terms in Eq. (5) are now known except I_5 .

Equation (5) is solved for $\dot{I}_5$.

I_5 is differentiated to obtain $\dot{I}_5$.

Since differentials are noisy, $\dot{I}_5$ is smoothed with a 1-msec time constant first order lag. This time constant is appreciably shorter than system response time.

The control system described in Ref. 29 produces the V_2 and V_3 (control voltages) used in Eqs. (2) and (3). V_2 controls I_1 ; V_3 controls R_1 .

Error currents e_2 and e_3 are formed around proportional and rate deviations of I_1 and R_1 :

$$e_2 = 4(\Delta I_1 + 0.005 \dot{\Delta I}_1) \quad (8)$$

$$e_3 = 60(\Delta R_1 + 0.005 \dot{\Delta R}_1) \quad (9)$$

From error currents e_2 and e_3 are derived setpoint currents for production of V_2 and V_3 . The relationship is a differential one:

$$\dot{I}_{c2} = \frac{e_2}{T_2} = \frac{e_2}{0.005} = 4(200 \Delta I_1 + \dot{\Delta I_1}) \quad (10)$$

and

$$\dot{I}_{c3} = \frac{e_3}{T_3} = \frac{e_3}{0.005} = 60(200 \Delta R_1 + \dot{\Delta R_1}) . \quad (11)$$

Integrating,

$$I_{c2} = 800 \int \Delta I_1 dt + 4 \Delta I_1 \quad (12)$$

and

$$I_{c3} = 12,000 \int \Delta R_1 dt + 60 \Delta R_1 . \quad (13)$$

Because of the form of the conversion between e and I_c , the proportional error signal now has an integral form, and the rate signal appears proportional. $\Delta I_2 = I_{2c} - I_2$ and $\Delta I_3 = I_{3c} - I_3$ are the final determinants of V_2 and V_3 .

The power supply chosen for the inner control winding has no capability for negative voltages. Its maximum voltage is +72 V. When $I_2 \geq I_{2c}$, V_2 is set at 0. When $I_{2c} > I_2$, V_2 is set at the smaller of 72 - 0.002 I_2 or $1000(I_{2c} - I_2)/I_{2c}$.

However, the power supply chosen for the outer control winding does have some capability for negative excursions. Its maximum positive voltage is +130 V. When $I_3 \geq I_{3c}$, $V_3 = -0.0028(I_3 - I_{3c})$. When $I_{3c} > I_3$, V_3 is set at the smaller of 130 V or $0.056(I_{3c} - I_3)$.

With plasma dynamics and the horizontal control system programmed on the analog machine, rapid determinations can be made of controllability under various operational upsets. The simulation is programmed to run at one-tenth reactor rate — a 6-sec excursion takes a minute to run.

(The slowdown is primarily for the benefit of the strip chart recorder; if runs become too lengthy, it could be eliminated.) Fast turnaround is important in control design in that it permits quickly repeated runs under slightly altered conditions, or on-line design studies.

With completion of the horizontal control model, examination of dynamic characteristics has just begun. Initial examination was for rate control only — no proportional. The test upset was a step in β_p from 1.21 to 1.31 and then back to 1.21. Results are shown in Fig. 9. R_1 and I_1 are remarkably well controlled; we see that the very simple system simulated, i.e., smoothed rate input, does a good job. Reduction

Fig. 9. Analog results for $\Delta\beta$ insertion and removal (steps); rate error signal only.

in β calls for negative V_3 , and, since the power supply is restricted in this polarity, resolution of this transient takes longer than for an increase in β . The largest excursion outside the control system is in I_5 , the symmetric eddy current in the liner. Increasing β_p produces a small increase in B_v , which in turn causes a sizable jump in I_5 . The impact of an increase in I_5 is to increase R_1 , and hence R_1 , which acts to reduce B_v . Thus, the eddy current I_5 is a built-in stabilizer for changes in β .

In the idealized machine model, rate control alone looks very good. However, in a practical case, rate signals are noisy, and putting in some proportional control as well is desirable. In the analog model, proportional control was first added for the control of R_1 through V_3 . The full amount called for in Eq. (13) was reduced to $I_{c3} = 5000 \int \Delta R_1 dt + 60 \Delta R_1$. Under these conditions, we obtained the outputs shown in Fig. 10. Initial control is poorer here than for rate only. (Note that the time per centimeter is ten times as great in this chart.) For an increase in β , the time to equilibrium has gone from 0.2 sec to about 1.5 sec; for a decrease in β , the time has gone from 0.5 sec to about 2.5 sec. Oscillations are clearly evident, and the excursions in V_3 are much greater, particularly in the $-\Delta\beta$ case. For the rate case, the limits are -16 and +20 volts; adding some proportional control requires swings from -45 to +100. Excursions in I_5 are about the same as before. Clearly, when adding proportional control, setpoints of rate and proportional signals must be carefully matched, time constants adjusted, and relative proportions optimized. The analog computer is well adapted to this process, and this will be the next task addressed.

There exists a digital batch program³¹ which is capable of much the same type of output as is our analog simulation. The turnaround on the analog is enormously faster, but, where differences exist in similar runs, the causes should be resolved. Figure 11 shows a digital run²⁹ for a ramp increase in β . In general, results are similar to our analog runs, but the digital curves show all current swings to be several times higher and faster. The I_5 excursion reaches 11.6 kA in the digital run vs 4.4 kA in the analog, and the baseline to baseline duration is 19 msec digital vs 50 msec in our Fig. 9. Within measurement error, the

Fig. 10. Analog results for $\Delta\beta$ insertion and removal (steps); rate error signal and proportional signal for ΔR .

areas under the analog and digital I_5 excursions are identical, indicating that the same magnitude of deviation is being compensated and that the analog feedback simply takes longer because of its smaller amplitude. Another task for future evaluation will be to make sure that inputs are the same in the two programs and to resolve existing differences in response.

Given satisfactory performance of the horizontal control system, incorporating both rate and proportional signals, integration of vertical

Fig. 11. Digital results for ramped increase in β .

control will be the next requirement. Interaction with the ISX-B project will provide that project with control design assistance and our project with an improved and tested model. Eventually, questions of startup, shutdown, and toroidal variations will need to be addressed, and these can be treated as extensions and modifications of our existing analog model of plasma dynamics.

5. MAJOR PLASMA DISRUPTIONS IN A TOKAMAK

5.1 INTRODUCTION AND BACKGROUND INFORMATION

Large fusion reactor systems used as commercial power producing devices must be reliable, maintainable, and economical and have a reasonable lifetime. Power tokamaks, as well as ignited devices such as TNS, must also be controllable — especially if the burn time is long enough to require fueling and helium removal. However, if the plasma in such a device goes into a disruptive instability during the burn cycle such that the plasma is lost in a very short time (100 μ sec-25 msec), the consequences to the first wall as well as to other components must be examined.

The reason for the concern is that a vast amount of thermal and magnetic energy is stored in the plasma. This energy is released in the disruption time τ_D to the wall and other components. Roughly 200 MJ of thermal and 200 MJ of PF energy are stored in the plasma. Two essential parameters for estimation of the damage to the first wall in particular are the disruption time τ_D and the nonuniformity of the surface heat flux impinging on the wall. Neither of these parameters can be definitively calculated, but the distribution of energy on the first wall is the more uncertain.

5.2 THE DISRUPTION TIME

The time scale of the major disruptive instability is not accurately known for TNS. A scaling law provided by Carreras et al.³² can be used to estimate the disruption time for TNS. The scaling law (ignoring plasma diamagnetic effects) for τ_D is proportional to $S^{2/5} \cdot \tau_R^{-1}$, where S is the ratio of the resistive skin time $\tau_R [= a^2/(\mu \bar{n})]$ and $\tau_{HP} [= R_o(\mu_o \rho_o)^{1/2}/B_{\phi o}]$. The ρ_o is the mass density, and $B_{\phi o}$ is the constant toroidal magnetic field. τ_{HP} is the time for Alfvén waves to propagate around the torus in the toroidal direction. This can be written as

$$\tau_D = C[\bar{R}^2 \cdot \bar{m}_i \cdot \bar{n}_i \cdot \bar{B}_{\phi o} \cdot \bar{a}^6(1/\bar{V}^3)]^{1/5} . \quad (14)$$

Table 1 (from Ref. 33) defines these quantities. C depends upon $q(r)$ and its derivative.

Also given in Table 1 are the values of the parameters in Eq. (14) for PLT and TNS. Notice that for TNS the disruption time is roughly 48 times that for PLT. The disruption time for PLT has been observed to be 500 μsec so that, if this scaling law is accurate, the projected disruption time for TNS is 0.024 sec. Figure 12, which is adopted from Ref. 32, shows this extrapolation for TNS.

5.3 LUMPED PARAMETER MODELS

A complete model for the time and space distribution of the ions and electrons during the disruption does not exist. The model of Ref. 34 does not completely describe the disruption process from beginning to end; therefore, a description of the plasma as it disrupts and strikes the first wall is desirable. In the analysis that follows, two models are used for the surface heat flux on the wall. The heat flux is assumed to be uniform in space but to have a time dependence as given below. The initial temperature of the wall was taken as 377K.

5.3.1 The Constant Velocity Model

Assume that the plasma has a parabolic spatial distribution in both density and temperature as shown in Fig. 13. Then, if the plasma moves radially outward as shown, the surface heat flux that strikes the wall as a function of time can be obtained.

5.3.2 The Gaussian Surface Heat Flux Model

The constant velocity model described above can be only an approximate formulation of the heat flux when the plasma disrupts, because all the heat energy does not go to the wall with all the particles clinging to the first wall. There will probably be a fair amount of heat energy left in the vacuum chamber even after the plasma disruption is over. In fact, if nothing were done, it would take some time before there was a thermal equilibrium between the remaining gas in the tokamak and the

Table 1. Parameters used in the disruptive scaling time for TNS

Symbol	Unit	Meaning	PLT	TNS
$\bar{R}_0$	Meters	Major radius	1.3	5
$\bar{m}_i$	Proton masses	Ion mass	2.5	2.5
$\bar{n}_i$	10^{13} cm^{-3}	Ion density	10	13
$\bar{B}_{\phi 0}$	Tesla	Toroidal field	5	5.3
$\bar{a}$	Meters	Minor radius	0.45	1.25
$\bar{V}$	Volts	Voltage at limiter	2-4	0.05-0.1

Fig. 12. Scaling law for obtaining the disruption time for TNS.

Fig. 13. Constant velocity model for the plasma disruption.

wall temperature. Therefore, it was decided that a Gaussian model would also be considered for the plasma disruption. This has two advantages. First, the heat deposition rate does not go to zero after a time t_D . Second, a comparison can be obtained between the two models (Gaussian and constant velocity) to see how sensitive the results are to the surface heat flux. Notice that at $t = 0$, i.e., the beginning of the disruption, the plasma has already started dumping some energy on the wall.

5.3.3 Consequences to First Wall

The first wall, being closest to the plasma, will be most directly affected by the plasma disruption. As the plasma current dies away and

the heat flux strikes the wall, eddy currents will be induced into the first wall, adding to the total heat load. These currents may have the beneficial effects of lessening the temperature gradient in the wall. Table 2 provides the results from the lumped parameter model; f_E is the fraction of the magnetic energy that goes into Joule heating.

5.4 DISTRIBUTED PARAMETER MODELS

A one-dimensional analysis of the temperature profile was done by means of the HEATING5³⁵ code. The temperature gradient in the wall during the plasma disruption was used to obtain the thermal fatigue resulting from the disruption. If the disruption time is 100 μ sec, some of the wall will melt and a portion of the wall will be vaporized.

5.4.1 Temperature Distribution in the First Wall

Three contributions to the heat load in the first wall are considered:

1. The surface heat flux impinging on the wall as the plasma disrupts.
2. Joule heating due to the eddy currents induced on the wall.
3. Neutron flux interacting with the stainless steel and causing a heat load internal to the wall. (The neutron flux heating contribution was terminated after the steady-state temperature distribution was achieved.)

The very high energy electrons (>1 MeV) can also contribute to the internal energy generation, but these are neglected. Also, any gamma rays (x rays) emanating from the plasma were neglected in these calculations.

5.4.2 Results and Observations

Figures 14 and 15 indicate the results of HEATING5. For the 100- μ sec case, ~ 4 -5 μ of the wall ablates and ~ 40 μ melts. For the 24-msec case, the wall neither melts nor ablates. In fact, the difference between the maximum temperature and the average wall temperature is about 407°C.

Table 2. Results of lumped parameter model ($T_i = 377\text{K}$)

Quantity	$\tau_D = 10^{-4} \text{ sec}$		$\tau_D = 2.4 \times 10^{-2} \text{ sec}$	
	Constant velocity	Gaussian	Constant velocity	Gaussian
Maximum temperature (K)	502	550	464	498
Time of maximum temperature (sec)	0.0135	0.0126	0.024	0.043
Maximum current (A)	5.31×10^6	5.30×10^6	1.65×10^6	1.65×10^6
Time of maximum current (sec)	4.54×10^{-4}	5.4×10^{-4}	0.0175	0.0175
Maximum Joule heating (W/m^3)	2.94×10^{10}	3.20×10^{10}	2.85×10^9	2.82×10^9
f_E	0.782	0.781	0.308	0.311

ORNL/DWG/FED 78-1212

Fig. 14. Temperature profile in wall for $\tau_D = 24$ msec, constant velocity surface heat flux and includes Joule heating.

ORNL/DWG/FED 78-1213

Fig. 15. Wall temperature profile for $\tau_D = 100$ μsec, constant velocity surface heat flux.

Thermal fatigue estimates for the stainless steel have been made, and about 460 aborts could be tolerated (less the fatigue due to the normal thermal cycling).

The maximum wall temperature for the 100- μ sec case was achieved at its surface at 70 μ sec and was $\sim 5400^\circ\text{C}$. The maximum depth of wall melting was 35 μ (if the latent heat of fusion is included in the calculation) and occurred 100 μ sec after the plasma disruption began. If the latent heat of fusion is ignored in the calculation, 40 μ of the wall melts.

The heat of vaporization cannot be conveniently included in the HEATING5 code; therefore, 100- μ sec runs are correct up to 3090K but not beyond that temperature.

One other fact to be noted is that the internal heat generation rate due to eddy currents changes the wall temperatures by roughly 1% of the temperature changes due to the surface heat flux. This can also be observed from the lumped parameter models. Therefore, the temperature distribution is not sensitive to the eddy current heat generation rate.

The 100- μ sec heat pulse causes ablation of material from the surface. Classical heat conduction may not be applicable to this situation because of the formation of a "virtual limiter," i.e., a cold plasma or sheath protecting the wall (to some extent) from the incident heat flux. This question should be investigated further.

5.5 THE NONUNIFORM ENERGY DEPOSITION PROBLEM

All the calculations to date have been done assuming a uniform energy deposition on the wall. However, several factors tend to make the deposition nonuniform. These include:

1. The center of the plasma will not be at the center of the first wall.
2. Because the vacuum vessel is toroidal, more heat will likely be deposited on the interior of the first wall.
3. The first wall will likely be D-shaped rather than circular. If the D is constructed of "flats," then this would even accentuate the nonuniformity.

4. Magnetic islands form in the plasma. As a result, regions of ergodicity are formed by the coupling of magnetic modes. The thickness of the ergodic region will determine the particle transport; thus, the energy deposition will depend upon the nonuniform thickness of these regions of ergodicity.
5. Magnetic field nonuniformities (such as TF ripple) will likely lead to nonuniformities in the energy deposition.

There may be factors of three to one in energy deposition due solely to the ergodic regions formed in the magnetic field lines. Quantification of these factors is an essential task that must be addressed.

5.6 WALL LIFETIME CONSIDERATIONS

The number of major plasma disruptions that can be tolerated before the wall fractures can be estimated from the temperature gradients obtained in HEATING5. The maximum thermal stress σ_{MAX} is given in Ref. 36.

Table 3 provides the value of these parameters for the disruption in TNS. The number of cycles permitted before failure was obtained from Section III (Nuclear Power Plant Components) of the ASME Boiler and Pressure Vessel Code.³⁷

5.7 CONCLUSIONS AND RECOMMENDATIONS

The following conclusions are derived from the analysis.

1. It is likely that the plasma disruption time will be longer than previously anticipated. Instead of 100 μ sec, the disruption time will probably be 24 msec, or even longer, for TNS.
2. For a τ_D of 24 msec, the temperature rise in the first wall (316 stainless steel) will be roughly 400°C; thus, about 450 disruptions could be accommodated before wall failure due to thermal fatigue. This calculation assumes a uniform energy deposition.

Table 3. Thermal stress limits on the first wall
(constant velocity model)

Quantity	τ_D	
	24 msec	100 μ sec
T_{MAX} ($^{\circ}C$)	588	5688 ^a
$\bar{T}$ ($^{\circ}C$)	181	759
ΔT ($^{\circ}C$)	407	4929
α ($^{\circ}C^{-1}$) @ $\bar{T}$	1.904×10^{-5}	2.071×10^{-5}
K ($1 \leq K \leq \infty$)	1	1
n ($0 \leq K \leq 1$)	1	1
E (psi)	26×10^6	26×10^6
a	1	1
μ	0.30	0.30
σ_{MAX}	2.88×10^5	3.79×10^6
S_a (ksi)	144	1895
N (number of cycles for fracture)	450	1

^aWall melts so that fatigue analysis no longer applies.

3. The eddy currents are much more important in force considerations, power supply protection, etc., than in contributing to the heat load through I^2R (Joule) heating. It seems that less than 10% of the temperature change in the wall will be due to the induced eddy currents.

4. The temperature rise is not very sensitive to the shape of the heat pulse that strikes the wall. The disruption time τ_D and the total thermal energy in the plasma determine the pulse height.

The following recommendations seem appropriate at this time.

1. Further study should be done on forces induced in coils, in the first wall, etc. Power supply protection must be made concrete so that a recommendation can be made concerning the best way to protect the OH supplies.

2. Further analysis should be done on the thermal consequences to the first wall. This analysis should be especially concerned with the nonuniform energy deposition case. Parametric studies can first be made; these should be done primarily for the 24-msec disruption time.

3. Initiation of control mechanisms could begin assuming the 24-msec case. It is even possible to start examining hardware requirements for gas puffing for first wall protection.

4. The preliminary results of this report should be improved by considering more detailed models (accounting for both latent heat of fusion and vaporization) and also improved and updated as MHD disruptive theory evolves.

5. Fusion devices currently under construction (such as Doublet III) should be analyzed in terms of consequences to the first wall with the idea of recommending experiments to be performed. For example, thermocouples could be put into the wall for temperature measurements, and attempts to measure the heat pulse and the nonuniformity of the disruption could be considered. This would be of direct concern not only for this analysis but also to theorists who study MHD disruptions.

6. R&D NEEDS ASSESSMENT

The determination of the needed R&D and the development of a "road map" depicting how the C&I development needs should be included require the following steps:

1. Characterize the end uses in detail.
2. Define the discrepancies between present capabilities and ultimate needs.
3. Define the R&D needs to fill the gaps in capability, and schedule within the bounds of funding expected.

It is quite clear that in the early stages of this study such findings may lack the desired accuracy. However, the general results are expected to provide valuable guidance by showing that the relative importance and the timely incorporation of experimental and calculational results from parallel fusion efforts will continue to improve the accuracy of the road map.

6.1 INSTRUMENTATION AND CONTROLS PRELIMINARY SPECIFICATIONS

6.1.1 Function

This system is required to control the tokamak automatically through all its phases of operation. Instrumentation is provided for controlling functions and for diagnostics. The data from the instrumentation are processed and displayed or stored in an archival system for later on-call display. The philosophy of operation is that of semiautonomous subsystem controls and diagnostics operating under the supervision of a central computerized control.

A significant emphasis in the development of this system is the necessary designs for fail-safe operations and degraded mode options.

6.1.2 Design Requirements

Operational

The system shall be centralized in the sense that one shall be able to turn on, operate, and secure the machine from a central location.

During each pulse of the machine, all the data relevant to the machine performance shall be collected in a central data pool for servicing all consoles provided for and used by experimenters and operators.

The device control shall be under the control of the central operations station. Substations or subsystems semiautonomous control stations may, upon direct assignment from the central control, operate a particular subsystem. Provisions will be made (by appropriate interlocks and overrides) so that individual subsystem operations such as testing and checkout will not affect other elements, operating or dormant. The responsibility to monitor all operations and the ability to override commands from substations shall remain with the central operations station.

Structural

The equipment shall be constructed with normal industry practice and modified as necessary for shielding from the detrimental effects of nuclear radiation and magnetic fields.

Configuration

The system shall consist of the central operating station and several substations. For ease in test and checkout, each substation shall be located, as the environment permits, in proximity to the equipment it is monitoring and controlling.

Maintenance

The equipment shall have redundancy, diversity, and fail-safe features to minimize the need for maintenance. The design life of the system shall be ten years with the exception of certain monitoring devices that, because of exposure to plasma operation, may require periodic replacement. Only monitoring devices and communications links and lines shall require remote maintenance.

6.1.3 Design Description

The system consists of four major divisions, each with its monitoring and data handling:

1. Auxiliaries control — controls and monitors the elements that are infrequently commanded, such as TF power system, vacuum pumping systems, refrigeration systems, electrical substation.
2. Emergency control — is the fail-safe, scram-type system that prevents machine damage from any significant anomaly.
3. Operations control — controls all machine operations during all phases of the burn cycle and during the immediate preparation for the next burn cycle. The supervisor or central station control and substation form this operations control.
4. Instrumentation monitoring — measures all pertinent variables and processes the data through appropriate algorithms for use.

The instrumentation for the emergency control shall receive the largest emphasis on redundancy and reliability. Monitoring of these variables shall be on at all stations.

The instrumentation for normal operation (control and monitoring) shall maximize the use of multiplexing microwave and light transmission to ease remote maintenance and to achieve an economy of space and penetrations from the operating stations to the machine.

The parameters which need to be sensed include:

1. Plasma density spatial and temporal variations,
2. Poloidal magnetic field strength and coil currents,
3. Toroidal magnetic field strength and coil currents,
4. Neutron flux at first wall,
5. Tritium levels at selected locations,
6. Neutral beam parameters and status,
7. RF system parameters and status,
8. Fueling rate,
9. Diverter temperature and coil currents, and
10. T_i and T_e of plasma — spatial and temporal variations.

6.2 STATUS OF THE SYSTEM AND EQUIPMENT DEVELOPMENT AND AVAILABILITY TO MEET THE NEEDS

The preceding section describes the present understanding of the instrumentation and controls needs of TNS/ETF. It is readily apparent that the controls needs for tokamak reactors are far from being well understood. The projected transient operation of a potential reactor has only recently been a subject for in-depth studies; thus, it seems to follow that a major benefit would result from a continuing in-depth investigation into defining the needs and potential solutions for TNS/ETF. The results from such studies will undoubtedly play a significant role in the early definition of potential problems and in the delineation and application of potential solutions which will shape the transient operation or burn cycle scenario for the fusion device.

6.3 REQUIRED R&D FOR INSTRUMENTATION AND CONTROLS

The major controls R&D requirement to meet the needs of TNS/ETF is the delineation and test of the physics and the controls concepts desired for one complete burn cycle. To date the tokamak controls R&D effort throughout the fusion community has been limited to very preliminary studies and tests of plasma position control. The major areas of control concerns of startup, burn, refueling, shutdown, and abort have hardly been considered. The approach to controlling a long burning tokamak has only recently been addressed in any significant fashion.

Thus, in Table 4, where needed R&D has been identified and scheduled against the background of the expected major TNS/ETF milestones, the conceptualization and study of the tokamak control options are, along with concerns listed as key subheadings, shown as continuing needs.

Appearing in those subheadings of instrumentation and controls needs are:

1. Determination of reactor-relevant instrumentation that can adequately survive the projected environment and provide the data for control.

Table 4. TNS/ETF R&D needs — instrumentation & controls

	78	79	80	81	82	83	84	85	86	87	88	89	90
TNS/ETF program schedule major milestones	Available development period (40 months) →					LPTT							
						Design Fabricate Install	OPERATE						
1. Define overall I&C needs including ^a Abort/shutdown Startup Heating Burn Refueling Plasma position						TNS							
						Title I	Title II	Bid	Award	Fabricate	Install	Check Out	OPERATE
2. Detail needs and define ^a solutions for: 1. Reactor-relevant instrumentation 2. Subsidiary systems 3. Safety-scrum systems 4. Diagnostic systems													
						Incorporate experimental results from ISX, PLT, TFTR, Doublet III, etc.							

^aThe activities should be continuing and will allow additional clarification of specific R&D needs as the studies progress.

2. Assessment of subsidiary (no direct interface with the plasma system) systems, their controls needs, problems, and solutions. Systems such as: heat removal, tritium handling, power supplies, etc.
3. Environmental systems to monitor and control the radioactive conditions.
4. Definition of the diagnostic systems approach and equipment matrix to provide the required support to experimenters and operators on both a real-time and an archival basis.

Table 4 represents the outlines of a tokamak instrumentation and controls development road map that are presently taking shape. The first application of the I&C system concepts being studied is proposed for the TNS machine (or an ETF with the same range of parameters) for a 1990 start of operations. This provides some useful bounds to the R&D cycle. The TNS machine will probably demand more sophistication in its I&C system than will the later DEMO and commercial machines. As envisioned the TNS/ETF program will, because of its development objectives, require more flexibility from the I&C system as operational variations are explored.

The preliminary TNS/ETF schedule shows that the necessary I&C developments for its operation must, in large part, be available by October 1983. It is deemed necessary to test out many of the I&C developments on a long pulse machine before TNS/ETF design starts. A machine such as the Long Pulse Technology Tokamak (LPTT) would be appropriate, and most developments must be ready for test on it. Some parallel developments may be permitted during the first few years of the project.

Increased definition of the I&C developments needed will become more apparent as studies continue. The R&D needs road map will be maintained current and will display these needs as they are defined.

7. I&C SYSTEMS DESCRIPTIONS

The understanding of what a tokamak operational scenario may be is slowly evolving, and, although the specifics of a needed control system are not known, the general requirements are sufficiently available to warrant the initiation of this study of potential control problems. The results of this study will influence the design of the tokamak in an effort to minimize these problems as much as possible.

The tokamak selected for study is one of the same general size and operation as the TNS device being evolved by the Advanced Systems Program at ORNL. Within the operating scenarios to be selected, there are five sequential phases or cycles to be followed: prebreakdown, breakdown (ionization of gas), startup, burn, and shutdown (see Fig. 1). One of the more difficult control problems during burn involves a plasma disruption (abort or unplanned shutdown) from unpredictable causes. These sequential steps are being described and the description kept current as well as the evolving understanding permits.

Before the discussion of these operational sequences, there is a discussion of the plasma parameters which must be measured and of the sensors which can possibly be used to detect these physical quantities. These measurements and the control actions which they exercise have to do with normal plasma behavior. The problems that may arise from abnormal plasma behavior may be very much more serious.

7.1 SENSING SIGNALS

Signals must be provided that are functions of the time variation of the plasma characteristics of temperature, density, position, and shape. There are undoubtedly other characteristics of the plasma that it will be helpful, if not necessary, to measure, but these four are essential. From the work to date, it appears that there are sensors which hold promise of giving satisfactory signals to be used for control if they can be developed to operate in the severe environments expected.

Frequency shift in scattered laser light may, with development, provide a means of determining both electron temperature and electron

density of the plasma. The bell-shaped curve of intensity of the scattered light can be analyzed to obtain the width at half maximum, and the area under the curve can be obtained. The first number (width) is proportional to temperature; the area is a function of the electron density.

Microwave absorption may also be used to measure plasma density. If a microwave sweep frequency generator injects a microwave beam into the plasma and if a receiver is positioned diametrically opposite the injection point, a null method of measurement may be used. As the frequency of the signal is swept over a band, there should be a microwave cutoff for a given plasma density and magnetic field. A signal proportional to the frequency at cutoff may be used to indicate plasma density.

Plasma location has been determined in existing devices by small inductive pickup loops positioned azimuthally around the plasma chamber. When the pulsed fields of the poloidal windings are bucked out, the voltages induced in these loops can be used to show plasma position in the chamber. Protection and cooling for these loops from an ignited plasma would require careful design, but this principle of sensing plasma position should be adaptable to a tokamak reactor.

Measurement of plasma current (average) has been done successfully with Rogowski coils. This method appears to provide a satisfactory means of measuring plasma current in an ignited reactor.

A qualitative evaluation of the electron temperature profile may be obtained by soft x-ray detectors mounted at different radii above the plasma. The intensity of the output from the photomultipliers will give some approximate idea of the location of the center of the plasma.

7.2 OPERATIONAL SEQUENCES

Within these boundary conditions of design and assumptions of sensors, the sequence of operating a pulsed tokamak, as presently being considered, from startup through one burn pulse and shutdown, is as follows:

When the desired preoperation pressure is attained, deuterium at some higher pressure is introduced. This pressure is low (on the order

of 10^{-4} torr). At this point, a low temperature, low density discharge is initiated. The number of such discharges necessary appears to be based on the overall characteristics of the discharge. A spectrographic analysis of the gas after the discharge may yield useful information on the effectiveness of the discharge cleaning.

After completion of the plasma chamber conditioning, the first step toward startup is to energize the bias winding on the transformer core, which is magnetized by this winding to approximately 10-15% below saturation as indicated by a fluxmeter measurement.

Next, the OH transformer is energized in opposite polarity to the bias winding with sufficient current to approximately cancel the magnetization of the bias winding. The OH driving system is now cocked, and the system is ready for the pulse initiation to begin the startup of the plasma. The gas within the torus is now established at the desired pressure of the proper species proportions of deuterium and tritium. Then the mechanical limiter is driven some 50 cm toward the plasma center. The purpose of this limiter is to enable a smaller radius plasma to be initiated. This smaller radius plasma will then be expanded (by ramping out the limiter) to the full operational volume of the plasma.

It is now time to begin initiating the plasma discharge. A pulse of microwave energy is introduced from the high field region (TF coil field) to ionize a slab of gas near the center. This pulse is on the order of 0.1-sec duration. After approximately 100 msec of microwave injection, the OH system voltage is ramped down from the cocked position and driven through zero and on to a reversed polarity voltage equal in magnitude to the original cocked potential. This ramped voltage swing is made in 2-5 sec.

As the voltage undergoes this swing, the mechanical limiter is withdrawn at a rate of approximately 50 cm/sec. This may be a programmed withdrawal, or it may be coupled to the plasma current. A Rogowski coil signal proportional to the plasma current can be used in one concept as a signal to the servo positioning the limiter. The scenario is designed to initiate a small plasma and, in a controlled manner, to expand it reasonably rapidly to the operating volume.

As the plasma current builds up and as the plasma volume is expanded, both plasma temperature and ion density must be maintained. Neutral beam injection is used to increase plasma temperature. Fuel addition by either pellet injection or gas puffing maintains the density as the plasma is expanded. Sensors of both temperature and density must provide signals to actuate the beam and fueling functions; closed control loops may be required for these operations.

As previously described, signals that are functions of plasma temperature and density are obtained by scattered laser and microwave absorption. The signal resulting from plasma density is used to control the fueling of the device in order to keep the plasma density constant. The plasma temperature derived from the laser signal corrected by the density signal from the microwave may be used to control the neutral beam injection to maintain the plasma temperature at the desired level.

This plasma temperature and density control is important during the brief (1- to 5-sec) plasma expansion phase. It may also be required, however, during the plasma burn because both plasma density and temperature may be subject to change as the long pulse of plasma burn continues.

When the plasma has been stabilized at the desired volume and has been brought to the necessary temperature and density, ignition takes place. Onset of ignition is determined by a rather marked increase in neutron emission from the plasma as indicated by a neutron flux detector. During the burn, the rate of rise of the OH driving voltage is greatly reduced, providing just enough rate of change to continue driving the plasma current. The signal for driving the OH transformer may be derived from the Rogowski coil which detects the primary plasma current. During the burn, the plasma position is monitored by the inductive pickup loops located azimuthally around the plasma. These loop signals are used to control the equilibrium field (EF) coil currents to maintain position of the plasma.

It is possible (but not yet determined) that the bias winding can be used to add some volt-seconds for increasing the burn time of the plasma. When the available volt-seconds for maintaining the burn have been exhausted, the voltage on the OH winding is ramped down to zero.

Additional factors must be considered for an orderly shutdown. One way to begin to turn off the plasma may be to stop fueling and let the reaction products build up so that excess "ash" gradually degrades the plasma. However, much more study is necessary to increase understanding of how slow extinction of the plasma can be assured.

When the plasma is extinguished, the torus volume must be pumped out to remove helium and high-Z impurities. At present, there are insufficient data to permit an accurate estimate of the plasma downtime necessary to prepare for the next plasma burn. Analyses of residual gas will be required to determine when conditions in the torus satisfy criteria for starting the plasma again.

Thus, control of a tokamak under normal conditions is conceivable although admittedly it presents formidable problems. However, abnormalities in operation pose a very different set of problems. Indeed, it is difficult even to define the possible kinds of such malfunctions and the types of problems they might present.

One possible problem might arise if the equivalent of a plasma runaway occurred. When ignition is attained, what maximum plasma temperature (particle energy) might result? Is there a self-limiting temperature maximum, and how certainly can it be predicted? Design of the first wall cooling system must accommodate this upper limit in a machine which investigates the physics of burning plasmas. Presently there seems to be some ambiguity as to the maximum plasma energy to be expected.

Probably the most serious condition which may arise is a sudden extinction of the plasma for some reason. How fast can the plasma current decay, and what voltage spikes would result on the windings linking the plasma? Would the eddy currents resulting in the TF coils cause quenching?

Where is the energy stored in the plasma dumped? If the plasma distorts asymmetrically and a large percentage of the plasma dumps on a relatively small wall area, it will undoubtedly result in a melt-through of the blanket wall. Protective measures and emergency actions that can cope successfully with such an event have not, at this time, been identified. This very serious possibility requires as much realistic evaluation as can be brought to bear on the problem. A significant effort, already

planned, will be devoted to the "abort" phase of operation. This problem is more closely considered in Sect. 5.

7.3 EXTERNAL SYSTEMS

There are several important systems external to the nuclear island which are necessary for the operation of a fusion power plant. Among these are the heat removal or cooling system, the very complex tritium handling system, the involved remote handling equipment, and the power generation system. There are, of course, others, but these principal ones have been studied to some degree to determine whether or not there are significant instrumentation and controls problems in these areas.

7.3.1 Heat Removal System

This system poses two problems: minimizing the thermal cycling of the blanket and providing thermal storage to accommodate the pulsed power of the reactor to the constant steam flow at constant temperature to the steam turbine.

Minimizing thermal cycling

Minimizing thermal cycling of the blanket involves control of the flow of helium coolant through the blanket, which has not been analyzed sufficiently to determine helium flow characteristics. Control possibilities are cutting off the coolant, retaining full helium flow, or modulating this flow in a controlled manner by a helium bypass valve. In any case, there are no instrumentation and controls problems anticipated in handling this problem successfully. The usual temperature sensors, flowmeters, pressure measuring devices, and gas throttling valves can be adapted for the required service. The need for new or exotic instrumentation is not anticipated in a successful control of the primary heat removal system.

Thermal Storage System

The use of a molten salt secondary coolant loop between the primary helium circuit and the steam system appears necessary. The loop performs two very necessary functions: thermal storage or thermal inertia to smooth out the power pulses, and a system in which tritium contamination of the helium system can be sequestered and continuously scavenged to prevent its entry into the steam system.

There is a wealth of experience at ORNL on the chemistry and handling problems associated with the use of molten salts. Pumps are literally off-the-shelf items, having undergone thousands of hours of trouble-free operation. Tight shutoff valves have not been used. However, valves with tolerable leakage rates have been used and will suffice for use in the molten salt loop. Level detectors, flowmeters, and pressure sensors have also been used extensively. In summary, there appear to be no instrumentation and controls problems associated with the use of a secondary salt coolant loop.

7.3.2 Tritium Handling

The handling of tritium presents one of the major problems attending the solution of a controlled fusion reactor based on magnetic confinement. One of the more difficult aspects of this problem concerns inventory of and accountability for the large amounts of tritium and its extreme mobility. Along with this problem is the propensity for tritium to exchange with normal hydrogen found in profusion in nature and, thus, to become chemically bound in such compounds.

Tritium in gaseous form can be volumetrically measured with additional tracking of radioactive decay by counting techniques. The gas, being a low energy beta emitter, must be passed directly through the counting chamber. Internals of the chamber would, therefore, need to be cleaned periodically because wall absorption will lead to serious errors. This cleaning is a nuisance, but obviously it is possible.

Large amounts of tritium are expected to be stored as hydride. Concentrations of large quantities can be determined by calorimetry

because beta emission will result in heat generation in the mass of hydride.

Because tritium exchanges so readily with hydrogenous material, it is important to choose carefully the insulators to be used in tritium instrumentation. Fluorocarbon compounds are to be avoided because fluorine presents a very undesirable contamination in tritium systems.

Although these problems exist, instrumentation for the tritium system does not appear to present any particular difficulties or the need for new invention in the way of instrumentation.

7.3.3 Remote Maintenance

It is far too early to attempt to evaluate the specific instrumentation required for effecting remote maintenance. There are, however, a few general conclusions that can be reached on the basis of what functions are known to be necessary.

Very little, if any, of the anticipated maintenance operations can be preprogrammed. Therefore, each operation must have visual observation, which will necessitate portable stereo television cameras.

Because of the "clutter" around some of the components requiring service, it will be necessary to penetrate this peripheral clutter to view the site. Fiber optics will almost certainly be required to gain visual access to some of these areas.

Load cells will be required rather generally. Because most of the maintenance is done by vertical lifts, the crane hooks must be instrumented to detect loads imposed by interferences and to avoid damage to restrained components.

7.3.4 Electrical Generation

The electrical generation system is similar in every way to any central station generation plant, and adequate instrumentation is available for this system.

7.4 REDUCED POWER CAPABILITY

Any power producing plant, even base load sized plants, must have the capability of being turned down to about 50% load. Preliminary considerations show that it should be possible to operate a tokamak successfully down to about 50% of design point power.

The exact means of accomplishing this need further clarification, but the major control knob appears to be fueling rate, which is effective (apparently) because the temperature of the plasma is a function of ion density squared. If the density is decreased by cutting down the fueling rate, the power in the plasma can be lowered satisfactorily.

In making changes in power level by decreasing fueling rate, it may be necessary to trim the control by bursts of neutral beam injection to keep the plasma temperature in a satisfactory operating regime.

8. PLANS FOR CONTINUING STUDY

The continuing activities are to be conducted within the framework of the six tasks previously outlined:

Task I — Prepare (and maintain current) the characterization of instrumentation and controls needs of future magnetic confinement fusion devices from experimental devices to power reactors. Define startup to shutdown/abort scenarios.

Task II — Prepare (and maintain current) the description of the physics to be controlled, the desired bounds for the variables, the most suitable parameters to be sensed for control and data handling, and the candidate sensor techniques to be considered.

Task III — Initiate the preparation of a control system hybrid computer model that will provide the "test bed" for the control approaches being considered and provide the insight into the evolution of optimum algorithms and control law selection.

Task IV — Maintain an integration with current tokamak experiments and utilize the pertinent results. Define experimentation that can be accomplished on present or near-term devices (e.g., ISX) and that will provide support and guidance to the theory-devised control laws.

Task V — Prepare (and maintain current) a development road map depicting the I&C R&D needs, time-phased, showing interfaces with device programs and indicating need dates and potential costs.

Task VI — Maintain communications with all pertinent developers, receive data on needs, results, and approaches, and provide status on this program's progress to all interested. Plan and organize appropriate presentations, workshops, and technical society participation.

The major emphasis continues to be on the determination of the desired operating scenarios as based on the physics requirements evolving from calculations and experiments. The future investigations are expected to include more study of the problems of disruptions, aborts, and shutdown

as well as the difficulties of defining sensors and instrumentation systems that can survive the reactor's environment and meet the accuracy and response demands of the controls and diagnostics.

Thus, the primary emphases for FY 79 are:

1. Definition of the physics/I&C relationships for the complete burn cycle.
2. Definition of the abort/hard shutdown problems and the search for solutions.
3. Hybrid computer I&C simulation correlation with the ISX-B experiment and expansion to long pulse calculations.
4. Study of the impact of the expected reactor environment on the instrumentation choices.

The secondary emphases will be on the continuing work in:

1. Updating the R&D road map.
2. Expanding the communications within the U.S. fusion community.
3. Studying the external (indirect relationship with plasma) systems problems.
4. Seeking correlation with tokamak experiments in addition to ISX-B.

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