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SYSTEMS CODE ASSESSMENT OF INNOVATIONS, MAJOR DESIGN DRIVERS, AND MINIMUM SIZES OF INTOR AND ETR-LIKE DESIGNS

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OAK RIDGE NATIONAL LABORATORY

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ABSTRACT

System studies of next-generation superconducting tokamaks are presented here. These studies include examining design changes suggested for the International Tokamak Reactor (INTOR) as a means of reducing the size and simplifying the device and assessing the impact of a series of more aggressive design assumptions suggested in recent Engineering Test Reactor (ETR) studies. Also, a set of candidate design points offering small machines (major radius - 4 m) with a relatively conservative mix of design assumptions is proposed. Some of the design assumptions found to have a major effect on the minimum size are TF coil current density, noninductive current drive, plasma elongation and edge q, plasma temperature for current drive, maximum allowable plasma beta, the minimum required wall load, and assumptions on fixed radial dimensions such as shield thickness, gaps, and plasma scrapeoff. Some design assumptions with less impact on the device size are the OH coil current density, PF configuration (limiter/divertor), and plasma current level.

1. INTRODUCTION

System studies have played a valuable role in many previous reactor-grade superconducting tokamak studies.¹⁻⁸ These studies generally resulted in large-size devices with major radii (R) > 5 m. However, recent concepts for possible improvements in designing components (such as superconducting magnets,⁹ rf current drive,¹⁰ and shielding design¹¹) offer the possibility of a smaller-size, next-generation machine such as the Engineering Test Reactor (ETR), which is currently being considered as a project to address physics and technological issues.

Our purpose is to evaluate the impact of these proposed design changes on an ETR-like device and identify modifications having the greatest influence. We have chosen the International Tokamak Reactor (INTOR)¹² as a candidate device on which the design changes are applied and have used a new version of the Fusion Engineering Design Center (FEDC) Tokamak Systems Code to carry out the investigation.

These studies are grouped into three categories: (1) design changes proposed by the INTOR Innovations Workshop¹³ as means of reducing the size and simplifying the device; (2) examination of the potential size reductions due to more aggressive improvements in the design assumptions, as suggested by recent ETR studies,¹⁴ which result in significant size reductions ($R < 3$ m); (3) some ETR-class designs which offer relatively small sizes [$3.8 < R$ (m) < 4.5] but which have relatively conservative design assumptions. The new version of the Tokamak Systems Code is described in the following section, and results of the studies are presented in Sect. 3.

2. MODEL

A new version of the FEDC Tokamak Systems Code has been created to study superconducting magnet tokamaks. The new model incorporates portions of the old systems code;¹⁵ uses improved models for several areas, such as superconducting current limits and poloidal field (PF) coil current scaling; and borrows the numerical optimization methods

from the new Compact Ignition Tokamak Systems Code.¹⁶ The new modeling procedures are briefly discussed here and are presented in more detail in the Appendix.

A primary improvement in the code is the incorporation of the optimization methods. For this purpose, the code is arranged to include a set of constraints that must be satisfied and a set of design parameters that are varied through the use of the optimizer, which also minimizes a figure of merit. (In this work the major radius is always minimized.) The primary "physics" constraints are a plasma density limit (Murakami or Greenwald), a beta limit (Troyon), a minimum wall-load limit, and an optional minimum ignition margin limit. The engineering constraints include superconducting-coil critical-current limits for the toroidal field (TF) coils and the ohmic heating (OH) or pusher coil; in-plane stress limits in the TF coil; minimum thickness requirements for the inner and outer shields; and stress limits for the bucking cylinder. Also, for the inductive-drive cases, the PF coil system flux capability is required to match the plasma volt-second requirement. An improved magnetohydrodynamic scaling routine is employed to calculate the PF coil currents as the design is varied. The superconducting-coil analysis, taken from ref. 17, is based on the Westinghouse Ni^3Si cable-in-conduit conductor design. Critical coil currents are computed with temperature and strain effects accounted for, and the TF coil in-plane stress is calculated with the assumption of bucking only. The minimum shield thickness requirements are calculated by using the subroutines from the old systems code (ref. 15), which include TF coil conductor nuclear heating rate, insulator dose rate, and biological exposure rate (for the outboard shield) effects. The bucking-cylinder modeling procedure follows the formalism from ref. 15 and includes stress limits for buckling and bearing pressure effects. All cases require the radial buildup of inboard components and gaps to be consistent with the major radius. These constraints are summarized in Table 1, along with a list of the variables used for iteration. Although the individual modeling procedures are global in nature, they include the primary considerations governing superconducting tokamak

Table 1. Constraints and variables used in the systems code modeling

Constraints	Related variables
Plasma density $\leq$ Murakami limit, Greenwald limit	Plasma density
Plasma beta $\leq$ Troyon limit	Plasma temperature
Wall load $\geq$ input value	Field on axis
Ignition margin $\geq$ input value ^a	Major radius
Plasma current $\geq$ input value ^a	Aspect ratio
TF coil current $\leq$ critical current	TF coil current density
OH coil current $\leq$ critical current	OH coil current density
TF coil in-plane stress $\leq$ allowable	TF conduit case thickness
Inboard, outboard shield thickness $\geq$ minimum requirement	Inboard, outboard shield thicknesses
Bucking-cylinder thickness $\geq$ minimum requirement (buckling and bearing stresses)	Bucking-cylinder thickness
PF coil flux capability $\geq$ plasma induction requirement ^a	OH coil thickness and bore
Major radius = summation of inboard component thicknesses	TF coil thickness

^aOptional constraint.

design. The purpose of this study is not to produce detailed design points but rather to investigate the trends that can be expected from changes in design choices.

3. RESULTS

The trade studies presented here are divided into three categories: (1) trade studies specified by the INTOR engineering group, (2) major sensitivity studies with broader ramifications for the device size, and (3) some intermediate trades with modest design choice changes. The case numbers referred to in this section correspond to those in Table 2, which summarizes the results. The first part of the

Table 2. INTOR systems code scoping studies--major cases

Options ^a	Case ^b					
	1 INTOR ref.	2 Opt. ^c INTOR	3 TFC size	4 D-null 1	5 D-null 2	6 SS low TT
<u>Design parameters</u>						
R (m)	5.00	4.84	4.84	4.94	4.44	4.32
a (m)	1.20	1.13	1.15	1.15	0.94	1.35
B (T)	5.5	5.6	5.6	5.6	5.5	4.5
$n_e (10^{14} \text{ cm}^{-3})$	1.6	1.7	1.7	1.7	1.6	1.5
Beta (%)	4.65	4.60	4.58	4.60	5.26	6.58
Beta-fast (%)	0.28	0.26	0.26	0.27	0.31	0.39
Q-I	2.24	2.18	2.25	2.21	2.26	2.07
P-DT (MW)	586	546	545	556	482	531
IM-KGH	0.86	0.88	0.85	0.87	0.91	0.81
IM-AsdexH	1.64	1.59	1.56	1.61	1.48	1.55
f-MVA	1.00	0.96	0.71	0.60	0.61	1.43
d-shield (cm)	80	80	80	80	80	80
d-TFC (cm)	67	69	68	63	62	102
d-bk-cyl (cm)	28	27	27	28	25	18
d-OHC/PC (cm)	54	44	44	49	46	32
Bore (cm)	115	113	115	123	102	31
Dose-TFC (Grad)	0.26	0.26	0.26	0.26	0.27	0.26
J-OHCp (kA/cm ²)	1.61	1.91	1.95	2.00	1.95	2.48
J-TFCp (kA/cm ²)	2.43	2.47	2.50	2.71	2.90	1.98
B-max-TFC (T)	11.1	11.1	11.1	10.9	10.8	11.1
<u>Design choices and options</u>						
I_p (MA)	8.0	7.7	7.7	7.7	7.1	10.4
$\frac{I_p}{q}$	2.26	2.26	2.26	2.23	2.26	2.26
T_e (keV)	10.0	9.7	9.7	9.9	10.0	10.0
g factor	3.84	3.84	3.84	3.84	3.84	3.84
κ	1.88	1.88	1.88	1.88	2.24	1.88
WL (MW/m ²)	1.17	1.17	1.17	1.17	1.17	1.17
Div/Lim	S-null	S-null	S-null	D-null	D-null	S-null
TFC-case (cm)	10	10	10	10	10	10
TFC ripple	0.012	0.012	0.024	0.012	0.012	0.012
CD method	Ind	Ind	Ind	Ind	Ind	rf
Burn time (s)	150	150	150	150	150	67
f-VS cap	1.06	1.00	1.00	1.00	1.00	0.29
Sol cap (Wb)	66	59	57	57	58	6.8
Scr.off (cm)	20	20	20	20	20	20
<u>Fraction of limits or constraints^d</u>						
f-beta	1.00	1.00	1.00	1.00	1.00	1.00
f- n_e	1.00	1.00	1.00	1.00	1.00	1.00
f-OHC-I/t	0.38	0.38	0.38	0.38	0.38	0.38
f-TFC-I/t	1.06	1.06	1.06	1.06	1.06	1.06
f-inner-sh	1.00	1.00	1.00	1.00	1.00	1.00
f-outer-sh	1.00	1.00	1.00	1.00	1.00	1.00
f-TFC-stre	0.86	1.00	1.00	1.00	1.00	1.00
f-buckling	1.00	1.00	1.00	1.00	1.00	0.51
f-tearing	0.68	0.72	0.69	0.67	0.67	1.00

Table 2 (continued)

Options ^a	Case ^b					
	7 SS high T	8 Ergodic edge	9 Limiter plasma	10 High I (10 MA)	11 High I (12 MA)	12 Improved OHC-1
<u>Design parameters</u>						
R (m)	4.89	5.38	4.81	4.97	5.41	4.65
a (m)	1.66	1.15	1.02	1.41	1.54	1.17
B (T)	4.7	5.8	5.6	4.8	5.2	5.4
n _e (10 ¹⁴ cm ⁻³)	0.78	1.51	1.69	1.39	1.40	1.70
Beta (%)	7.44	5.02	4.98	5.73	4.60	4.90
Beta-fast (%)	1.50	0.41	0.32	0.39	0.28	0.27
q-I	2.00	1.92	1.86	2.15	2.17	2.25
P-DT (MW)	729	675	488	673	791	531
IM-KGH	0.80	0.95	0.95	0.85	0.97	0.83
IM-AsdexH	1.71	1.84	1.58	1.78	2.01	1.54
f-MVA	--	0.67	0.47	1.20	1.50	0.84
d-shield (cm)	80	80	80	80	80	80
d-TFC (cm)	112	61	56	52	59	73
d-b'-cyl (cm)	23	31	28	26	30	24
d-OHC/PC (cm)	35	63	62	44	48	48
Bore (cm)	38	132	118	119	134	88
Dose-TFC (Grad)	0.26	0.26	0.26	0.26	0.26	0.26
J-OHCp (kA/cm ²)	2.27	1.94	1.99	1.90	1.79	2.43
J-TFCp (kA/cm ²)	1.66	2.74	3.18	3.55	2.88	2.40
B-max-TFC (T)	11.4	10.8	10.6	10.2	10.7	11.2
<u>Design choices and options</u>						
I _p (MA)	15.0	8.4	7.4	10.0	11.9	8.0
q	2.26	2.27	2.23	2.26	2.26	2.26
T _e (keV)	20.0	11.5	10.4	10.5	10.0	9.6
g factor	3.84	3.84	3.84	3.84	3.07	3.84
κ	1.88	1.88	1.88	1.88	1.88	1.88
WL (MW/m ²)	1.17	1.17	1.17	1.17	1.17	1.17
Div/Lim	S-null	Limiter	Limiter	S-null	S-null	S-null
TFC-case (cm)	10	10	10	10	10	10
TFC ripple	0.012	0.012	0.012	0.012	0.012	0.024
CD method	rf	Ind	Ind	Ind	Ind	Ind
Burn time (s)	227	150	150	150	150	150
f-VS cap	0.34	1.00	1.00	1.00	1.00	1.00
Sol cap (Wb)	10	69	56	66	85	54
Scr.off (cm)	20	40	20	20	20	20
<u>Fraction of limits or constraints^d</u>						
f-beta	1.00	1.00	1.00	1.00	1.00	1.00
f-n _e	0.56	1.00	1.00	1.00	1.00	1.00
f-OHC-I/t	0.38	0.38	0.38	0.38	0.38	0.86
f-TFC-I/t	1.06	1.06	1.06	1.06	1.06	1.06
f-inner-sh	1.00	1.00	1.00	1.00	1.00	1.00
f-outer-sh	0.99	0.99	1.00	1.00 ^e	1.00	1.00
f-TFC-stre	1.00	1.00	1.00	0.96 ^e	1.00	1.00
f-buckling	0.42	1.00	1.00	1.00	1.00	1.00
f-bearing	0.97	0.63	0.60	0.63	0.67	0.77

Table 2 (continued)

Options ^a	Case ^b					
	13 Improved OH-2	14 WL, shield	15 SS, D-null	16 Adv. TFC	17 Scr.off snield	18 TIBER phys.
<u>Design parameters</u>						
R (m)	4.28	4.28	3.56	2.91	2.34	2.89
a (m)	0.95	1.00	1.06	0.89	0.73	1.02
B (T)	5.3	5.5	4.3	4.4	4.5	4.81
n _e (10 ¹⁴ cm ⁻³)	1.81	1.60	1.03	1.13	1.22	1.07
Beta (%)	5.59	4.48	8.05	8.45	8.57	11.4
Beta-fast (%)	0.33	0.26	1.08	1.14	1.16	2.65
q-I	2.30	2.28	2.13	2.10	2.09	1.80
P-DT (MW)	471	327	331	234	144	380
IM-KGH	0.90	0.75	0.74	0.64	0.49	0.61
IM-AsdexH	1.46	1.18	1.09	0.82	0.54	0.77
f-MVA	0.52	0.63	0.58	0.62	0.61	1.78
d-shield (cm)	80	65	65	63	51	52
d-TFC (cm)	65	63	91	57	48	59
d-bk-cyl (cm)	22	24	14	23	16	18
d-OHC/PC (cm)	43	43	32	17	18	27
Bore (cm)	89	99	17	11	11	14
Dose-TFC (G-rad)	0.27	1.5	1.6	4.9	27.7	8.5
J-OHCp (kA/cm ²)	2.74	1.96	2.04	3.26	3.32	3.45
J-TFCp (kA/cm ²)	2.85	2.91	2.60	2.69	3.73	2.74
B-max-TFC (T)	10.8	10.9	10.5	12.1	11.7	12.2
<u>Design choices and options</u>						
I _p (MA)	7.4	6.4	9.5	8.6	7.3	13.8
q	2.26	2.26	2.26	2.26	2.26	2.26 ^f
T _e (keV)	9.9	9.8	15.0	15.0	15.0	24.6 ^f
g factor	3.84	3.84	3.84	3.84	3.84	4.05
κ	2.24	1.88	2.24	2.24	2.24	2.18
WL (MW/m ²)	1.17	0.90	0.90	0.90	0.90	1.44
Div/Lim	D-null	S-null	D-null	D-null	D-null	D-null
TFC-case (cm)	10	10	10	3.5	3.5	3.5
TFC ripple	0.024	0.012	0.012	0.012	0.012	0.012
CD method	Ind	Ind	rf	rf	rf	rf
Burn time (s)	150	150	SS	SS	SS	SS
f-VS cap	1.00	1.00	0.23	0.22	0.21	0.29
Sol cap (Wb)	55	45	1.5	0.5	0.5	2.1
Scr.off (cm)	20	20	20	20	4.7	4.7
<u>Fraction of limits or constraints^d</u>						
f-beta	1.00	1.00	1.00	1.00	1.00	1.00
f-n _e	1.00	0.84	0.59	0.52	0.43	0.25
f-OHC-I/t	1.00	0.38	0.19	0.17	0.18	0.45
f-TFC-I/t	1.06	1.06	1.06	2.88	2.88	2.88
f-inner-sh	1.00	1.20	1.20	1.33	1.66	1.48
f-outer-sh	1.00	1.00	1.00	1.00	1.00	1.00
f-TFC-stra	1.00	1.00	1.00	1.00	1.00	1.00
f-buckling	1.00	1.00	0.55	0.05	0.05	0.19
f-bearing	0.70	0.69	1.00	0.67	0.67	0.87

Table 2 (continued)

Options ^a	Case ^b					
	19 TIBER II	20 QSS, IM=0.86	21 Ind. OH, IM=0.86	22 QSS, $\kappa=1.88$	23 S-null, $\kappa=1.88$	24 D-null, IM=1
<u>Design parameters</u>						
R (m)	3.00	3.78	4.03	4.38	4.52	4.12
a (m)	0.83	0.95	0.87	1.20	1.29	1.14
B (T)	6.0	4.9	5.5	5.1	4.9	4.8
n_e (10^{14} cm^{-3})	1.17	1.79	1.90	1.61	1.54	1.71
Beta (%)	8.07	6.53	5.39	5.59	5.80	7.32
Beta-fast (%)	1.88	0.373	0.300	0.358	0.369	0.56
q-I	1.98	2.23	2.31	2.14	2.15	2.18
P-DT (MW)	318	399	392	556	605	738
IM-KCH	0.50	0.856	0.856	0.856	0.856	1.0
IM-AsdexH	0.65	1.33	1.32	1.64	1.70	1.95
f-MVA	0.79	0.58	0.56	0.76	1.31	0.81
d-shield (cm)	52	80	80	81	80	82
d-TFC (cm)	50	71	66	73	79	72
d-bk-cyl (cm)	6	17	22	22	21	19
d-OHC/PC (cm)	40	29	39	35	34	32
Bore (cm)	55	63	87	84	74	70
Dose-TFC (Grag)	8.6	0.27	0.27	0.26	0.26	0.27
J-OHCp (kA/cm^2)	6.37	4.09	3.01	3.55	3.10	3.68
J-TFCp (kA/cm^2)	2.93	2.87	2.84	2.53	2.33	2.72
B-max-TFC (T)	12.5	10.7	10.8	11.0	11.1	10.8
<u>Design choices and options</u>						
I_p (MA)	10.0	7.9	6.7	9.0	9.6	10.5
$\frac{I_p}{q}$	2.27	2.26	2.26	2.26	2.26	2.26
T_e (keV)	24.6 ^f	9.8	9.7	10.3	10.3	11.2
g factor	4.05	3.84	3.84	3.84	3.84	3.84
κ	2.18	2.24	2.24	1.88	1.88	2.24
WL (MW/m^2)	1.44	1.17	1.17	1.31	1.25	1.69
Div/Lim	D-null	D-null	D-null	D-null	S-null	D-null
TFC-case (cm)	3.5	10	10	10	10	10
TFC ripple	0.012	0.012	0.012	0.012	0.012	0.012
CD method	rf	QSS	Ind	QSS	QSS	QSS
Burn time (s)	SS	301	150	386	410	427
f-VS cap	0.25	0.57	1.0	0.63	0.65	0.59
Sol cap (Wb)	6.0	25	51	32	34	33
Scr.off (cm)	4.7	10	10	10	20	10
<u>Fraction of limits or constraints^d</u>						
f-beta	1.00	1.0	1.0	1.0	1.0	1.0
f- n_e	0.26	0.96	0.96	0.95	0.97	1.0
f-OHC-I/t	3.63	1.0	1.0	1.0	1.0	1.0
f-TFC-I/t	2.88	1.06	1.06	1.06	1.06	1.06
f-inner-sh	1.49	1.0	1.0	1.0	1.0	1.0
f-outer-sh	2.86	1.0	1.0	1.0	1.0	.99
f-TFC-strs	1.01	1.0	1.0	1.0	1.0	1.0
f-buckling	--	1.0	1.0	1.0	1.0	1.0
f-bearing	--	0.85	0.72	0.82	0.90	0.85

^aSee Table 3 for definitions of symbols and abbreviations.^bSee Table 4 for brief descriptions of the cases.^cAt OH coil I/t limit for both beginning of pulse and end of burn.^dRelative to INTOR or FEDC assumptions.^eTF coil case thickness at 2.5 ms.^f $T_e = 16.5 \text{ keV}$.

table lists some of the calculated parameters, and the middle section lists the specified design choices which delineate the cases from each other. Finally, the bottom section lists the fraction of the constraint limits reached. Table 3 describes the parameters listed in Table 2, and Table 4 briefly explains the design choice changes introduced in each case.

3.1 INTOR TRADES

3.1.1 INTOR Reference (Case 1)

Case 1 is the systems code representation of the INTOR Phase Two A, Part 2, design.¹² Major parameters, such as the radial build thicknesses, toroidal field, and plasma density and temperature, are held fixed. The plasma profile factors are adjusted here so that the fusion power matches that specified for INTOR. The results from the systems code analysis were compared with the INTOR specifications, and the following calibrations were performed so that several quantities are set to their limits as stated in the INTOR reference.¹² The g factor in the beta limit is set to 3.84 and the Murakami density limit coefficient is set to 1.46 so that the beta and density limits are just met. The calculated TF coil dose rate and biological exposure rate are used as maximum allowable values so that the shield thicknesses are at the minimum values. Also, the upper bound of the TF (OH) coil current per turn is set to 1.06 (0.38) times the calculated allowable value by using the FEDC analyses described in the Appendix. The FEDC analyses for bucking-cylinder stress and TF in-plane stress are retained as described in the Appendix without any calibration. The calculated wall load of 1.17 MW/m^2 is used in the INTOR cases as the physics requirement. We note that the PF coil flux capability is 6% larger than the plasma volt-second requirement ($f\text{-VS} = 1.06$) for this case, when the complete double-swing OH coil current is utilized. The single-null reference PF set used for scaling purposes in case 1 is shown in Fig. 1.

Table 3. Definitions of parameters of the
INTOR systems code scoping studies

Symbol	Definition
<u>Design parameters</u>	
R	Plasma major radius
a	Plasma minor radius
B	Vacuum toroidal field at plasma major radius
n_e	Volume-averaged electron density
Beta	Volume-averaged plasma pressure/ B^2
Beta-fast	Volume-averaged energy density of the slowing-down fusion alphas/ B^2
q-I	Cylindrical safety factor
P-DT	Power due to plasma D-T fusion
IM-KGH	Ignition margin assuming Kaye-Goldston confinement scaling modified for the H-mode
IM-AsdexH	Ignition margin assuming ASDEX H-mode confinement scaling
f-MVA	Poloidal field coil peak MVA requirement relative to the INTOR Phase Two A, Part 2, design reference (case 1)
d-shield	Inboard shield thickness
d-TFC	Inboard toroidal field coil thickness in the radial direction
d-bk-cyl	Toroidal field coil bucking-cylinder thickness
d-OHC/PC	Thickness of ohmic heating solenoid or inboard pusher coil
Bore	Clear bore radius of the above coils
Dose-TFC	Radiation dose at the inboard toroidal field coil leg through life of device
J-OHCp	Winding-pack current density in the ohmic heating solenoid or the pusher coil
J-TFCp	Winding-pack current density in the inboard leg of the toroidal field coil
B-max-TFC	Peak magnetic field at TF coil superconductor (T)
<u>Design choices and options</u>	
I_p	Plasma current
$\bar{q}$	Mean-field safety factor at edge of divertor plasma
T_e	Plasma electron temperature

Table 3 (continued)

Symbol	Definition
<u>Design choices and options (continued)</u>	
g factor	Coefficient for the Troyon beta limit scaling
κ	Elongation of plasma cross section
WL	Average neutron wall load
Div/Lim	Poloidal field configuration (divertor/limiter)
TFC-case	Toroidal field coil case thickness
TFC ripple	Toroidal field ripple at the plasma outboard edge due to the discrete coil legs in the absence of ferromagnetic shield pieces
CD method	Current drive method
Burn time	Flat-top time for plasma fusion burn
f-VS cap	Poloidal flux capability as fraction of the required flux for inductive current drive through a complete pulse
Sol cap	Induction flux capability of the ohmic heating solenoid
Scr.off	Inboard plasma scrapeoff thickness
<u>Fraction of limits or constraints</u>	
f-beta	Plasma beta as fraction of the design beta limit
f-n _e	Plasma electron density as fraction of the design density limit
f-OHC-I/t	Current per turn as fraction of its design limit in the ohmic heating or pusher coil
f-TFC-I/t	Current per turn as fraction of its design limit in the toroidal field coils
f-inner-sh	Inboard nuclear shield thickness as fraction of the design requirement
f-outer-sh	Outboard nuclear shield thickness as fraction of the design requirement
f-TFC-strs	Average in-plane stress in the toroidal field coil inboard leg as fraction of the design allowable
f-buckling	Buckling stress of the bucking cylinder as fraction of the design allowable
f-bearing	Bearing stress of the bucking cylinder as fraction of the design allowable

Table 4. Brief descriptions of 24 cases of the
INTOR systems code scoping studies

Case	Description
1 INTOR Ref.	Systems code reproduction (benchmark) of INTOR Phase Two A, Part 2, design. Design limits on beta, n_e , inner shield thickness, and outer shield thickness adjusted to INTOR design. Design limits on superconducting-coil current per turn (winding-pack current density), average in-plane stress at inner leg of TF coil, and bucking-cylinder buckling and bearing stress based on FEDC criteria. Current density in solenoid at end of burn substantially less than at beginning of pulse.
<u>INTOR studies</u>	
[Cases 3 through 13 assume the same limits and constraints as those in the optimized INTOR device (case 2) unless otherwise stated.]	
2 Optimized INTOR	Minimum-radius INTOR adjusted from case 1. Current per turn in solenoid at end of burn raised to value at beginning of pulse. In-plane stress of TF coil inboard leg raised to FEDC limit. Remaining INTOR or FEDC design limits in analysis still applied.
3 TF coil size	Minimum-radius design with TF coil size decreased to reduce stored-energy and hence peak MVA requirement in PF coil system. Ripple due to smaller TF coil alone allowed to double from case 1 (ferromagnetic shield segments assumed to be placed in shadow of coils to roughly halve the apparent field ripple at outboard edge of plasma).
4 Double null, $\kappa = 1.88$	Minimum-radius INTOR with $\kappa = 1.88$ elongation PF reference set. Outer PF coils close to midplane; favors vertical access. Stored energy similar to standard single-null case except for reduced MVA.
5 Double null, $\kappa = 2.24$	Minimum-radius INTOR with $\kappa = 2.24$ elongation PF reference set. Larger vertical separation of outer PF coils; horizontal access possibilities. Device size smaller because of higher κ . MVA and stored-energy levels over single-null case.
6 Steady state, low temperature	Minimum-radius, single-null, $\kappa = 1.88$ INTOR. PF set not required to provide volt-second capability. Temperature allowed to vary; settles at ~ 10 keV. Offers one of the largest size reductions (11%) of suggested INTOR trades.

Table 4 (continued)

Case	Description
7 Steady state, high temperature	Same as case 6 except plasma temperature is fixed at 20 keV to reduce steady-state rf power required for current drive.
8 Ergodic edge	Minimum-radius, $\kappa = 1.88$, limiter plasma INTOR. Limiter PF reference set used since poloidal divertor is not used for impurity control. Inboard and outboard scrapeoff widths doubled to account for additional area required for perturbed edge region of plasma.
9 Limiter plasma	Minimum-radius, $\kappa = 1.88$ INTOR using a limiter reference set.
10 High plasma current (10 MA)	Minimum-radius, $\kappa = 1.88$, single-null INTOR. Plasma current at least 10 MA.
11 High plasma current (12 MA)	Minimum-radius, $\kappa = 1.88$, single-null INTOR. Same beta as optimized reference case (case 2); g factor in beta limit equation lowered 20% to test sensitivity of minimum radius to uncertainties in plasma physics scaling.
12 Improved OH coil 1	Minimum-radius, $\kappa = 1.88$, single-null INTOR. OH coil current per turn limit set to FEDC allowables and ripple limit doubled (see case 3). First concept integration case.
13 Improved OH coil 2	Minimum-radius, $\kappa = 2.24$, double-null INTOR. FEDC OH coil allowable current per turn; ripple limit doubled (see case 3). Second concept integration case.

Major sensitivity studies

14 Wall load, shield change	Minimum-radius, $\kappa = 1.88$, single-null INTOR. Inner shield decreased to 0.65 m (by allowing thickness to be 20% smaller than calculated minimum thickness) to account for possible improvements in shield design. Wall-load requirement reduced to 0.9 MW/m^2 .
15 Steady state, double null	Minimum-radius, $\kappa = 2.24$, double-null INTOR. Same changes as introduced in case 14. Volt-second capability removed; current drive assumed. Plasma temperature fixed at 15 keV to lower rf power needs. FEDC OH coil current per turn allowable used.

Table 4 (continued)

Case	Description
16 Advanced TF coil	Minimum-radius INTOR. Same as case 15 except: outer TF coil case thickness reduced from 0.1 m to 0.047 m; current per turn in TF coil allowed to go to limit found by running a fixed TIBER comparison case (case 19).
17 TIBER radial build	Minimum-radius INTOR. Same as case 16 but uses TIBER radial-build lengths. Inboard scrapeoff is 4.7 cm (vs 20 cm for INTOR), first wall is 5 cm (vs 1.4 cm for INTOR), gap between shield and cryostat is 1 cm (vs 0.3 cm for INTOR), and inboard shield is 51 cm (vs 65 cm used in case 16), which is arrived at by allowing the shield to go to 66% over the minimum thickness.
18 TIBER physics	Minimum-radius INTOR. Same as case 17 but uses: TIBER plasma profile coefficients 0.5 for density (vs 0.27 for INTOR) and 1 for temperature (vs 0.54 for INTOR); g factor in beta limit of 4.05; Greenwald density limit; plasma current of at least 10 MA; wall load of at least 1.44 MW/m^2 ; $\bar{q} = 2.26$; $\kappa = 2.18$; triangularity of 0.55 (vs 0.35 for INTOR); same plasma temperature as case 19. (Same fluence as TIBER.)
19 TIBER fixed point	System code reproduction (benchmark) of TIBER II. Radial build held fixed. Results from models in systems code used as calibration for TIBER allowables in other runs.
20 Quasi steady state, 0.856 K-G ignition margin	Minimum-radius, $\kappa = 2.24$, double-null INTOR. Edge $\bar{q}$ of 2.26 (same as INTOR reference). Kaye-Goldston H-mode ignition margin of at least 0.856 [same as for INTOR reference (case 1)]. Minimum 1.17-MW/m^2 wall load. OH coil volt-second capability of at least 0.3 times total plasma volt-second requirement (for startup capability for quasi-steady-state scenario). Reduced inboard scrapeoff of 0.1 m (since this is double-null plasma).
21 Fully inductive, 0.856 K-G ignition margin	Minimum-radius INTOR. Same as case 20 except for full volt-second capability of PF set for inductive drive.
22 Quasi steady state, $\kappa = 1.88$	Minimum-radius INTOR. Same as case 20 except elongation is lowered to 1.88 (same PF reference set as used in case 4).

Table 4 (continued)

Case	Description
23 Single null, 0.856 K-G ignition margin	Minimum-radius INTOR. Same as case 22 except for single-null PF set and inboard scrapeoff thickness of 0.2 m specified for single-null operation.
24 Double null, 1.0 K-G ignition margin	Minimum-radius INTOR. $\kappa = 2.24$, double null, and Kaye-Goldston ignition margin of 1.0. Otherwise, same design choices as case 20: quasi steady state with OH coil providing 0.3 times the total flux requirement; FEDC OH coil current per turn allowables; and 0.1-m inboard scrapeoff.

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Fig. 1. INTOR reference PF system. (a) Single null; (b) limiter, $\kappa = 1.6$; (c) double null, $\kappa = 2.24$; (d) double null, $\kappa = 1.83$.

3.1.2 Optimized INTOR Reference (Case 2)

Case 2 is the minimum-major-radius optimized version of INTOR. It is slightly smaller than the reference case because the OH coil flux capability is fully utilized ($f\text{-VS} = 1$) and the TF coil in-plane stress reaches the FEDC limit, but overall it is similar to the INTOR reference (case 1). The optimized INTOR cases discussed in Sects. 3.1.3-3.1.8 should be compared with this case.

3.1.3 Reduced TF Coil Size (Case 3)

The first trade issue addressed is that of reducing stored-energy and MVA requirements by introducing ferromagnetic material between the outer TF coil legs and the plasma. The ferromagnetic inserts reduce the magnetic ripple at the plasma and allow the outer TF coil legs to move closer to the plasma, thereby reducing the PF coil sizes. Although the system code does not account for ferromagnetic-material ripple reduction per se, the issue is analyzed by allowing the calculated ripple to increase by a factor of 2. This factor seems to be a reasonable guess, based on previous design work on ferromagnetic inserts.¹⁸ The minimum-major-radius device found for this case is summarized in case 3 of Table 2. The basic device parameters are very similar to the optimized INTOR reference (case 2), the main exception being a reduction in MVA requirements of about 24% (and a similar reduction in stored energy). The outer TF coil leg has moved in from 9.23 m to 8.73 m, and the main part of the MVA reduction is due to the smaller associated inductance and required current in the outermost vertical field coil. All MVA and stored-energy numbers presented here are for the pulsed requirements (i.e., the PF coils and not the TF coils).

3.1.4 Double Null (Cases 4 and 5)

The impact of going to a double-null divertor system is addressed in cases 4 and 5, which are also minimum-major-radius cases. Case 4 uses a κ (elongation) = 1.83, double-null-divertor reference PF set (indicated in Fig. 1) for scaling purposes. Otherwise, the design choices for the reference case are used except for a slightly different edge $\bar{q}$. The

resultant device is similar to the optimized INTOR reference except that the MVA and stored-energy requirements are about 35% lower. As indicated in Fig. 1, the PF set for this device has a large vertical separation between the outer PF coils which could allow horizontal access. (The scaled PF sets found by the systems code have characteristics similar to reference sets from which they are derived.) The 2.24-elongation, double-null PF set shown in Fig. 1 is the reference set for case 5. The minimum radius is 4.44 m, and the size reduction is due primarily to the favorable scaling of plasma current with increasing elongation and to the increase of the critical beta with plasma current. This case has a 14% larger beta than any of the previous cases. Even though the elongation is almost 20% higher in this case relative to the optimized INTOR reference, the plasma height is actually slightly lower because of the sharply reduced minor radius. The MVA and stored-energy requirements are 36% and 46% smaller, respectively, than the optimized reference values. As a sensitivity test, this case was run with the edge $\bar{q}$ fixed at 3.0. With this modification, the resultant machine is similar to cases 2 and 4, with a major radius of 4.75 m, a plasma current of 7.6 MA, and a minor radius of 1.15 m. This similarity indicates that the edge $\bar{q}$ has a large impact on the minimum size. Also, these double-null-divertor cases use the same inboard scrapeoff length (0.2 m) as the single-null device. If the inboard scrapeoff length is halved (to 0.1 m) for double-null operation, additional major-radius decreases of about 0.2 m can be achieved.

3.1.5 Noninductive Current Drive (Cases 6 and 7)

Cases 6 and 7 evaluate the impact of using noninductive-current-drive methods to determine how much size reduction is possible when the OH coil is not required to provide the required plasma flux. Case 6 is a minimum-radius, 1.88-elongation, single-null INTOR with no volt-second constraint on the PF system. There is a 0.52-m reduction of the major radius, which is due primarily to the reduction in the OH coil size. Also, the aspect ratio has decreased from 4.3 to 3.2, and the magnetic field has dropped from 5.6 T to 4.5 T, with a resulting net increase in plasma current. In this run, the plasma temperature, which is allowed

to vary, settles at about 10 keV. Higher plasma temperatures tend to lower rf power requirements for current drive. To test the sensitivity of minimum radius of noninductive-current-drive devices to different plasma temperatures in case 7, the plasma temperature is held fixed at 20 keV; otherwise, the design choices are the same as for case 6. With the higher temperature, the minimum radius is 4.89 m. The large increase in radius is due to the inefficient use of plasma beta (i.e., operating above the peak in $\langle \sigma v \rangle / T^2$), which makes it more difficult to reach the minimum wall-load requirement. Even with no volt-second requirement imposed in cases 6 and 7, the PF set does provide about one-third of the flux requirements. Most of the flux provided in these cases (about 80%) is from the shaping and vertical field coils.

3.1.6 Ergodic Edge Divertor (Cases 8 and 9)

The ergodic edge layer has been proposed as a means of impurity control. This concept requires a larger plasma edge region than the conventional poloidal divertor, along with some additional coils to create the ergodic region. To address the impact of an ergodic divertor on the overall machine, we incorporate the following changes in the design choices for case 8. First, the limiter PF coil set shown in Fig. 1 is used as a reference for the PF coil scaling routine. Also, the scrapeoff-layer lengths are doubled from the reference values of 0.2 m inboard and 0.1 m outboard to account for the additional edge region space requirements. The minimum major radius is about 0.5 m larger than that of the optimized reference case, the increase being due primarily to the larger inboard scrapeoff width. Also, the MVA and stored-energy requirements are about one-third lower than the optimized INTOR reference because of the simpler PF coil set. For comparison, a minimum-radius INTOR limiter with no ergodic edge is shown in case 9. Here the same edge conditions are used as for the INTOR reference, and the device parameters are similar to the optimized reference single-null-device values except for the much lower MVA and stored-energy requirements.

3.1.7 High Plasma Current and Beta Sensitivity (Cases 10 and 11)

The use of a minimum plasma current as a plasma performance criterion has been suggested as an alternative to the use of an ignition margin based on some confinement scaling. The parameters of a minimum-radius device with a minimum 10-MA plasma current, in addition to the INTOR reference design choices, are shown in this case (case 10). A rather modest major-radius increase of 0.13 m is found for the 10-MA device. As with the noninductive-current-drive examples, the aspect ratio and the toroidal field both drop relative to the optimized reference case, and a net increase in the plasma current results. Case 11 examines the sensitivity of the results to uncertainties in the beta limit scaling. Here the beta value is required to be at least as large as in the optimized reference case (4.6%), and the g factor in the beta limit scaling is reduced 20%. This device also uses the reference 1.88-elongation, single-null, INTOR PF set for scaling purposes. The increase in the minimum radius to 5.41 m is much larger than that of the preceding 10-MA case, and a plasma current of 12 MA is therefore necessary to satisfy all the constraints. Most of the radial-build growth occurs in the OH coil and the plasma minor radius in case 11. As is generally true for the higher plasma current devices, the PF system MVA and stored-energy requirements are more severe, as can be seen by comparing cases 10 and 11 with case 2 (the 7.7-MA reference).

3.1.8 Concept Integration (Cases 12 and 13)

Cases 12 and 13 evaluate the improvements possible from a combination of new design concepts. In case 12, the allowable OH coil current per turn is raised from 0.38 times the FEDC limit (determined from the baseline INTOR comparison of case 1) to the FEDC allowable value to investigate the effects of using more advanced OH coil superconducting designs. Also, the allowable TF coil ripple is doubled (as in case 3). Case 12, which is for the 1.88-elongation, single-null PF set, has a 0.2-m decrease in minimum radius and a 13% decrease in MVA and stored-energy requirements as compared with case 2. Case 13 introduces the same design choice changes to the 2.24-elongation,

double-null divertor (case 5). The relative changes in minimum radius and in MVA and stored-energy requirements are similar to those for the single-null systems (cases 2 and 12) just compared. The minimum radius decreases 0.16 m, the drop in minimum size being primarily accounted for in the reduced OH coil size, which is possible with the increased current density. The MVA and stored-energy reductions are similar to those seen in comparing case 3 with case 2 and are due to the reduced outer TF coil radius.

3.1.9 INTOR Trade Study Summary

All the INTOR trade studies discussed previously were analyzed by optimizing for the minimum major radius (R). On the basis of this figure of merit alone, the sensitivities are small (<10%), the largest impact observed being an 11% decrease in R for the noninductive-current-drive devices. However, other quantities of importance were observed to be more sensitive. For instance, the MVA and stored-energy requirements can be reduced by one-third to one-half with different PF systems, such as the double null and limiter, even with little or no reduction in R. Figure 2 shows the minimum major radius and required MVA for some of the INTOR trades of cases 1-13.

None of the cases were ignited under Kaye-Goldston H-mode scaling; the baseline INTOR had an ignition margin of only 0.86. In cases 1-13, the smaller-R examples generally have even lower Kaye-Goldston ignition margins. For ASDEX H-mode scaling, all cases are ignited, and the sensitivity of ignition margin to size reduction is smaller than that for Kaye-Goldston scaling.

3.2 MAJOR SENSITIVITY TRADES

The following trade studies introduce design choices that result in larger major-radius reductions than those for cases 1-13. The design choices introduced in cases 14-19 are serial in that previous changes are also included in the following cases and represent a movement in parameter space from INTOR¹² to a device similar to TIBER II (Tokamak Ignition/Burn Experimental Reactor).¹⁴

Fig. 2. Summary of INTOR scoping studies.

3.2.1 Improved Shield, Reduced Wall Load (Case 14)

All of the INTOR trades presented in Sect. 3.1 assumed that an 0.80-m shield inner thickness is required for the INTOR specified neutron fluence. This assumption was based on the INTOR Red-Book analysis,¹² but improved shield designs may offer reduced thickness requirements. Also, longer burn times (which would occur in noninductive-current-drive schemes) could reduce the wall-load requirements to reach the specified fluence. For examination of the

impact of these possibilities, case 14 uses a 0.65-m inner shield and requires a wall load of only 0.9 MW/m^3 . The thinner shield thickness is reached in the computation by allowing the actual thickness to be 17% smaller than the minimum shield thickness based on the case 1 calibration (i.e., the upper bound on $f\text{-inner-sh}$ is 1.2). Otherwise, the design choices are the same as for the single-null INTOR reference. This case has a minimum major radius of 4.28 m, smaller than any of the preceding cases (cases 1-13). The 0.56-m size reduction relative to case 2 is about equally due to each of the two design choice changes introduced here. Also, the MVA and stored energy are reduced by one-third. This device results in the lowest plasma current (6.4 MA) of any of the devices.

3.2.2 Steady State, Double Null (Case 15)

In case 15 the PF system volt-second requirement is removed to simulate noninductive current drive, and the 2.24-elongation, double-null-divertor PF reference set used in case 5 is used for PF coil current scaling. Also, since higher plasma temperatures are desirable for noninductive current drive, the plasma temperature is fixed at 15 keV here. As in cases 12 and 13, the FEDC limit for the OH coil superconducting current per turn is used here, but this design choice has little impact since noninductive current drive is assumed. The minimum radius drops to 3.56 m, the aspect ratio drops from 4.28 to 3.36, and the plasma current increases to 9.5 MA. The large increase in plasma current is understandable since no volt-second constraint or limitation on current-drive power is imposed here. If the lower hybrid model of ref. 19 is used, the steady-state current drive requirement for this case is 96 MW.

3.2.3 Improved TF Coil (Case 16)

The TIBER II concept¹⁴ incorporates design choices different from those for INTOR¹² and generally represents less conservative approaches. Two primary differences exist in the TF coil: (1) a "TIBER II" allowable current per turn and (2) a thinner outer TF coil

case. The higher allowable TF coil current per turn limit is found by calibrating the TF coil model to the TIBER II reference¹⁴ in the same fashion described for INTOR⁹ for case 1. With these comparison values, the TF coil current per turn is allowed to go to 2.88 times the FEDC calculated limit, and the outer case thickness is reduced from 0.1 m to 0.035 m. Case 16 incorporates these changes in addition to those included in case 15, and there is an additional 0.65-m decrease in radius (a drop of 18%). The MVA, stored energy, and plasma characteristics of this device are similar to those of case 15. Although improved TF coil properties are introduced here, the TF coil winding-pack current density is only slightly larger than that for the preceding case. This is understood by realizing that the peak field in the superconductor is 12.1 T whereas it was 10.5 T in the preceding case. The TF coil current density is a more sensitive trade item than the OH coil current density. As in case 12, a similar level of improvement in the OH coil critical current as used here produces a major-radius reduction of less than 5%. This trend is confirmed when we double the OH and TF coil allowable critical currents (individually) for the optimized INTOR reference (case 2) and observe a 3% size reduction with OH coil improvements and a 6.5% size reduction with TF coil improvements.

3.2.4 Radial-Build Component Thicknesses (Case 17)

Some additional design choice differences between TIBER II and INTOR are the fixed radial-build component thicknesses. Case 17 introduces assumptions from ref. 14 for gaps and first-wall, shield, and scrapeoff thicknesses. The primary difference is a reduction of the 0.2-m inboard scrapeoff thickness used in the INTOR devices to 0.047 m. The shield thickness is allowed to drop to 0.51 m by letting the thickness be 40% lower than the FEDC calculated minimum required thickness (f -inner-sh = 1.66). Also, the number of TF coils is raised from 12 to 16, as used in ref. 14, but this change has little bearing on reducing the machine major radius. This case results in a minimum size of 2.34 m, which is the smallest of any of the cases examined in this study. Along with the small size is a fusion power of only 144 MW and a Kaye-Goldston H-mode

ignition margin of only 0.49, also the smallest of any encountered in this study.

3.2.5 TIBER II Physics (Case 18)

The TIBER II modeling used in ref. 14 is more closely approximated by introducing the following design choices. First, the plasma current is required to be at least 10 MA. Also, the beta-limit β coefficient is raised from 3.84 to 4.05; the Greenwald density limit is used instead of the Murakami limit; the minimum wall load is raised to 1.44 MW/m^2 (as calculated in the TIBER II comparison in case 19); the plasma density and temperature profiles are assumed to be more peaked than those used for INTOR; the elongation is lowered to 2.18; the plasma triangularity is raised from 0.32 to 0.55; and the total neutron fluence is set equal to the level of ref. 14. These changes raise the minimum radius to 2.89 m. Case 18 has a smaller aspect ratio than TIBER II because the OH coil in this case has no real purpose whereas TIBER II includes a pusher coil in the OH coil radial location which must provide a plasma indentation. This design choice difference is also reflected in the higher OH coil current density for TIBER II. When the aspect ratio in case 18 is forced to be the same as in case 19, minimum radius increases only slightly, to 3.0 m, and the plasma current, beta, and magnetic-field levels are also similar to those of the TIBER II comparison case (case 19).

3.2.6 TIBER II Comparison (Case 19)

In case 19, the radial build and plasma parameters are held fixed to those of TIBER II in order to calibrate the models used here to the design basis used in ref. 14. This is done in the same manner as for INTOR in case 1. These calibrations were used in cases 14-18. From these cases, we see that the reduced major radius of case 19, compared with the INTOR reference, is due primarily to: (1) smaller radial-build thickness of fixed components (e.g., scrapeoff and TF coil outer-case thickness); (2) improved TF coil superconductor properties (higher allowable current densities); (3) improved shield design (thinner

allowable shields); and (4) use of noninductive current drive (which removes the necessity of providing a large OH coil). Figure 3 shows the progression of major radius and magnetic field of cases 14-19.

3.3 POTENTIAL INTEGRATED POINT DESIGN OPTIONS

On the basis of previous cases, some potentially high-leverage items have been identified which reduce the required device size. In this section some representative potential options for a revised INTOR-type device using the design changes noted earlier are presented. The distinguishing design choices for these cases are ignition margin, PF configuration and elongation, and the current-drive method.

3.3.1 Kaye-Goldston 0.856 Ignition Margin, Quasi Steady State (Case 20)

Although the major sensitivity design choices introduced in Sect. 3.2 result in significantly smaller devices, the associated ignition margins are generally smaller than the INTOR design choice ignition margins. In case 20, we require a Kaye-Goldston ignition

Fig. 3. Impact of design changes in R vs B space.

margin of 0.856 [which is the value calculated for the INTOR reference (case 1)], while introducing some of the major sensitivity design choices. The allowable OH coil current per turn is set to the FEDC limit, which is about three times as high as the INTOR (case 1) level and about one-third that permitted in case 19 (the PF superconducting material of ref. 14 is different from that used in this analysis). To simulate quasi-steady-state conditions, the volt-second constraint is arranged so that the OH coil must provide 0.3 times the total volt-second requirement of the plasma. The reason for this is that a substantial fraction of the plasma flux requirement might have to be provided inductively for startup and should have volt-second control independent of the equilibrium field coils. The 2.24-elongation, double-null PF system is used here, with an edge $\bar{q}$ of 2.26. Since the double-null divertors require less inboard scrapeoff, the inboard scrapeoff width is lowered to 0.1 m. Otherwise, the INTOR reference design choices are used, and the resulting minimum radius is 3.78 m. As dictated by the volt-second constraint, the OH coil provides 25 Wb out of the total requirement of 84 Wb, which corresponds to a 300-s burn time in the quasi-steady-state operation mode. When the rest of the PF system volt-second contributions are included, the flux capability is almost 60% of the total requirement. To provide 70% of the full plasma flux requirements, the machine must grow by an additional 7 cm. The relatively large OH coil winding-pack current density is a reflection of the lower peak field in the OH coil (9 T) compared to the devices discussed in the following sections [cases 21-24 (9.5-10 T)].

3.3.2 Kaye-Goldston 0.86 Ignition Margin, Inductive OH Coil (Case 21)

As noted for the preceding device, even though the OH coil provides only 0.3 times the required plasma flux, the total PF system provides nearly twice that amount. In case 21, we investigate the impact on the minimum radius when a totally inductively driven system is required in addition to the design choices used in case 20. Only a 0.25-m increase in major radius is needed to accommodate the inductive flux requirement. This growth is accounted for primarily by a 0.34-m

increase in the OH coil thickness, which is needed to provide the additional flux. The plasma minor radius shrinks by 0.08 m, resulting in a higher aspect ratio, lower plasma current device with slightly lower total flux requirements than case 20.

3.3.3 Double Null, Low Elongation (Case 22)

This case examines the impact of using lower elongation. The only design change relative to case 20 is that the elongation is lowered to 1.88 (with the same PF reference set used in case 4). As previously noted, there is a strong sensitivity of the minimum radius to the elongation, and in this case the radius increases 16%. The sensitivity is related to the plasma current dependence on elongation and the close linkage of plasma current to the critical beta. Reductions in elongation tend to reduce the plasma beta, and size increases are required to recoup plasma performance equivalent to that with the higher elongation. In case 22, the 0.86 Kaye-Goldston ignition margin is a more severe limitation than the minimum wall-load requirement, as the resultant wall load is 12% higher than the minimum. In the previous two cases, both the ignition margin and the wall-load limits are met.

3.3.4 Single Null, Low Elongation (Case 23)

Case 23 is similar to case 22 except that the double-null PF set has been replaced with the 1.88-elongation, single-null INTOR reference (case 1) PF set. Accordingly, the inboard scrapeoff width is increased to 0.2 m. The minimum radius for case 23 is 4.52 m, compared to 4.38 m for case 22, and the size increase is due primarily to the doubled inboard scrapeoff width. The only design choice differences between this case and the optimized INTOR reference (case 2) are: (1) the higher allowable OH coil current per turn, (2) the quasi-steady-state flux requirement, and (3) the Kaye-Goldston ignition requirement of 0.86. With these changes included, the minimum radius is 0.32 m, smaller than the optimized INTOR reference of case 2.

3.3.5 Double Null, Kaye-Goldston 1.0 Ignition Margin (Case 24)

To determine the sensitivity to required ignition margin, we show the minimum-radius, 2.24-elongation, double-null device with a Kaye-Goldston ignition margin of 1 in case 24, which otherwise uses the same design constraints described in case 20. A 9% increase in major radius is needed to accommodate the 16% increase in required ignition margin. This increase in radial build occurs primarily in the OH coil and the minor radius. The fusion power has increased 85% compared with that of case 20. This increase is caused mainly by the increased plasma volume and, to a lesser amount, by the increased beta. This device has a wall loading of 1.69 MW/m^2 , which is much higher than the 1.17-MW/m^2 lower limit. When the ignition margin is increased to 1, the wall load becomes a much less stringent plasma requirement compared with the lower ignition margin devices. The sensitivity of this device to more volt-second capability is similar to that of case 20. When 70% of the full plasma flux must be provided by the PF set, the major radius grows by 8 cm to 3.86 m.

4. SUMMARY

A series of trade studies on superconducting tokamaks such as ETR and INTOR has been carried out. The trades are motivated by a combination of the scoping studies proposed by the INTOR group and innovative ideas suggested for ETR applications. The results of these studies can be summarized in two major categories, trades that have relatively minor impact on the reduction of the device radius and trades that produce major size change sensitivities. Table 5 lists the trades examined that have a small impact on changing the minimum size along with the cases that investigated these assumptions. Among the trades investigated here that result in small size reductions are doubling the allowable TF coil ripple, changing the PF configuration from single null to double null or to a limiter, increasing the plasma current requirement, and increasing the OH coil critical current density by a factor of 2. Some of the major sensitivity trades and assumptions identified in this study are listed in Table 6. These include

Table 5. Assumptions and design trades with small impact on the minimum major radius

Assumption	Change	Impact on major radius	Case examining the assumption
Ripple limit	Double allowable ripple	<1%	Case 3
PF configuration	Change from single to double null, to limiter	<2%	Cases 4 and 9
Plasma current	Increase requirement by 30%	3% Increase	Case 10
OH coil critical current	Increase by more than two times	4% Decrease	Cases 12, 13, and 16

Table 6. Assumptions and design trades with major sensitivities for size reduction

Assumptions	Change	Impact on major radius	Case examining the assumption
Fixed radial-build thicknesses	Decrease by 10% of R	20% Decrease	Case 17 (also see case 8)
TF coil critical current	Increase by almost three times	18% Decrease	Case 16
Noninductive current drive	Eliminate OH flux requirement	11-13% Decrease	Cases 6 and 15
Plasma elongation	Increase from 1.9 to 2.2	10% Decrease	Case 5 and 15
Required wall load	Reduce from 1.2 MW/m ² to 0.9 MW/m ²	9% Decrease	See case 14 discussion and case 18
Edge $\bar{q}$	Increase from 2.3 to 3.0	7% Increase	See case 5 discussion
Beta limit coefficient g	Decrease from 3.9 to 3.1	12% Increase	Case 11
Plasma temperature for current drive	Increase from 10 to 20 keV	13% Increase	Case 7

increasing the plasma elongation from 1.9 to 2.24, decreasing the operational plasma temperature from 20 to 10 keV, minimizing the OH coil by assuming noninductive current drive, and improving the TF coil critical current density by over a factor of 2. These high-leverage items should be considered in future superconducting tokamak design studies and in the comparison of results from different studies that use different assumptions.

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REFERENCES

1. K. Borass and M. Soll, Nucl. Eng. Des./Fusion 4, 21 (1986).
2. R. L. Reid and D. Steiner, Nucl. Technol./Fusion 4, 120 (1983).
3. R. L. Reid and Y-K. M. Peng, Nucl. Technol./Fusion 5, 356 (1984).
4. F. Rosatelli and G. Raia, Fusion Technol. 8, 362 (1985).
5. H. M. Attaga, G. A. Emmert, J. F. Santarius, and G. L. Kulcinski, Fusion Technol. 8, 367 (1985).
6. R. Whitley, D. Berwald, and J. Gordon, Fusion Technol. 8, 741 (1985).
7. D. Ehst et al., Fusion Technol. 8, 727 (1985).
8. For further references see: C. C. Baker, G. A. Carlson, and R. A. Krakowski, Nucl. Technol./Fusion 1 (1981).
9. J. Miller, L. Summers, and J. Kerns, Fusion Technol. 10, 1305 (1986).
10. D. Ehst et al., Nucl. Fusion 26, 461 (1986).
11. D. Smith and D-K. Sze, Fusion Technol. 10, 599 (1986).
12. International Tokamak Reactor Phase Two A, Part 2, STI/PUB/714 (ISBN 92-0-131086-2), Int. Atomic Energy Agency, Vienna, 1986.
13. Tokamak Concept Innovations, Report of a Specialists' Meeting on Tokamak Concept Innovations, IAEA-TECDOC-373, Int. Atomic Energy Agency, Vienna, January 1986.
14. C. Henning and B. G. Logan, eds., TIBER II Tokamak Ignition/Burn Experimental Reactor 1986 Status Report, UCID-20863, Lawrence Livermore Natl. Lab., 1986.
15. R. L. Reid et al., The Tokamak Systems Code, ORNL/FEDC-84/9, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., 1984.
16. J. Galambos, D. Blackfield, Y-K. M. Peng, R. L. Reid, D. J. Strickler, and E. Selcow, Systems Studies of Compact Ignition Tokamaks, ORNL/FEDC-86/5, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., 1987.

17. S. Kalsi, Nucl. Eng. Des./Fusion 4, 37 (1986).
18. C. Flanagan, D. Steiner, and G. Smith, Initial Trade and Design Studies for the Fusion Engineering Device, ORNL/TM-7777, Union Carbide Corp. Nuclear Div., Oak Ridge Natl. Lab., 1981.
19. A Second Stability Power Reactor, ANL/FPP/86-1, Argonne National Laboratory, 1986.
20. N. Uckan and J. Sheffield, A Simple Procedure for Establishing Ignition Conditions in Tokamaks, ORNL/TM-9722, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., 1985.
21. D. J. Strickler, Y-K. M. Peng, and J. D. Galambos, "Poloidal Magnetics of Compact Ignition Tokamaks," Bull. Am. Phys. Soc. 31, 1564 (1986).
22. Superconducting Magnet Coils for the Large Coil Program, Phase 2 of Final Report to Union Carbide Corp. Nuclear Div., Westinghouse Electric Co., Oak Ridge Natl. Lab. Contract No. 22X-31747C, March 1980.

APPENDIX

A.1 MODEL DESCRIPTIONS

Most of the modeling procedures used in the present superconducting version of the systems code are borrowed from previous works and integrated to form the new code. Rather than present the detailed equations of the models, we only summarize them here, for the most part, and reference the original sources.

A.2 PHYSICS

The present systems code uses the same profile-averaged global physics model used in the CIT optimization code.¹⁶ This model is also described in ref. 20. The density and temperature profile shapes are of the general form

$$f(x) = (1 - x^2)^\alpha, \quad (\text{A.1})$$

where x is the normalized plasma radius r/a ; a is the plasma minor radius; and α is the adjustable profile coefficient.

The fusion power is found by numerical integration over the plasma volume. In addition to the fusion power, the globally averaged bremsstrahlung power, ohmic power, and transport losses are included in the ignition margin calculation. The ignition margin is defined as the sum of power source terms (fusion and ohmic heating) over the sum of the power loss terms (transport and radiation). The confinement scalings used here for Kaye-Goldston and ASDEX H-mode are the same as those given in ref. 16 except that Kaye-Goldston confinement reported here refers to H-mode scaling and is actually two times that given in ref. 16 for the Kaye L-mode scaling. The wall load is defined as the total fusion power divided by the wall area, which is approximated as

$$A = 2\pi^2 R(a + SO)^2 \kappa, \quad (\text{A.2})$$

where R is the major radius, SO is the average scrapeoff width, and κ is the plasma elongation.

The beta is defined as the sum of the average plasma pressure ($\langle nT \rangle$) plus the fast-alpha pressure over $B^2/2\mu_0$. The fast-alpha pressure is calculated as 0.66 times the fast-alpha density times the average energy of the slowing-down spectrum. The fast-alpha density is proportional to the fusion power times the slowing-down time, and the fast-alpha energy is related to the critical slowing-down energy level at which the energy is deposited to the ions.

The plasma current is calculated with a scaling routine that is based on the average safety factor formalism for divertors.²¹ This procedure involves estimating the poloidal path length around the separatrix and is dependent on the plasma elongation and triangularity. The PF coil current scaling routine is the same as that used in ref. 16, and the coil inductance calculation (used for calculating volt-second capability) is taken from ref. 15. This procedure scales the PF coil locations and currents from a reference case that is generated from a magnetohydrodynamic equilibrium calculation. The cylindrical q ($q-I$ in Table 2) calculation uses the equation in ref. 12. The plasma flux requirement calculation is described in ref. 16, but no sawteeth effects are included here in the resistive losses during the burn.

The plasma electron density calculation includes contributions from a 50-50 deuterium-tritium fuel mixture, from an assumed 5% alpha particle ash population, and from impurities of mass number 8 which contribute 0.5 to the Z_{eff} .

The standard formulas for the Troyon beta limit and Murakami density limit, as listed in ref. 12, are incorporated here. The Murakami parameter used in these cases is taken to be 1.5, and the g factor in the beta limit is listed in Table 2.

A.3 SUPERCONDUCTING COIL ANALYSIS

The superconducting-coil analysis is taken from ref. 17. This model is based on a Westinghouse Ni^3Si CCIC design that was used in the Large Coil Program (LCP).²² Figure A.1 is a schematic of the superconducting conduit. The superconducting strands are in the center region of area A. In all cases presented here, the area A and its void fraction

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Fig. A.1. TF coil conduit geometry assumed in the model.
 Symbols: t_w is the conduit case thickness; t_i is the insulator thickness.

used for helium coolant are kept constant at the Westinghouse design values of 2.93 and 0.32 cm², respectively. The insulator wrap thickness t_i is fixed at 0.65 mm for all cases here; the conduit case thickness is held at 3.6 mm for the OH coil, and it is iterated for the TF coil (with a lower bound of 2.5 mm) so that the in-plane stress is satisfied, as described later. The critical current per turn given in ref. 17 for this conductor is used as the starting point for the calculation and is indicated by the upper curves in Fig. A.2. This current level is subsequently reduced for strain and temperature effects as described in ref. 17. The winding and cooldown strain levels are taken to be 0.007; the emf-induced strain calculation is explained subsequently. Also, the peak coolant temperature is assumed to be 4.7 K. This procedure gives the allowable critical current per turn, which is constrained to be less than or equal to the required current per turn. For the TF coil the required current per turn is calculated

Fig. A.2. Critical current levels used in the superconducting analysis.

by dividing the current in each coil (needed to produce the field on axis) by the number of conduits per coil. The number of conduits per coil is calculated by dividing the area per coil available to conduits by the individual conduit area. To find the area of the TF coil available to conduits, we take the total TF coil cross-sectional area as described in ref. 3 and subtract the area occupied by the TF coil external case (which is assumed to be a constant thickness, as specified in Table 2, around the TF coil periphery). Also, an additional 5% of the resulting TF coil area is assumed to be unavailable for conduits because of the impracticalities of completely filling up the coil with conduits.

The in-plane TF coil stress in the coil conduit case is evaluated by using the model of ref. 17. The coils are assumed to be bucked only, and the Tresca stress is required to be less than two-thirds of the yield or one-third of the ultimate, where the Tresca stress is the sum of the radial compressive component and the vertical tensile stress component. The radial component is calculated by dividing the centering force per TF coil by the sum of the radial cross-sectional area of the conduit case. The tensile stress is calculated by dividing the dilational force by the sum of the coil outer case and conduit case areas, where it is assumed that both are made of the same material. The allowable case stress used in these studies is taken to be 460 MPa. The emf-induced strain in the conduits is calculated by using the calculated stress level and a Young's modulus of 2×10^{12} Pa.

The OH coil modeling is similar to that of the TF coil. The same conduit used in the TF coil is used in the OH coil except that conduit case thickness is held fixed at 3.6 mm (since the OH coil stress is not modeled accurately). Also, an external case 3.5 cm thick is assumed to surround the entire OH coil. The OH coil strain is estimated by using the stress model described in ref. 16 to calculate the emf-induced strain; the Young's modulus is the same as that for the TF coil.

A.4 SHIELDING

The neutronics analyses are based on the model from ref. 15. The shield thickness requirements are based on limiting dose values to the TF coil (as listed in Table 2), maximum heating rates to the windings, and maximum shutdown dose rates (for the outer shield only). For results presented here, the first criterion always dominates. The routine considers a 90% steel, 10% water mixture for the inboard shield material and an 80% steel, 10% boron carbide, 10% water mixture for the outboard shield. Also, shielding effects from the first wall and TF coil outer case are considered. For the outer shield, a correction factor to the required shield thickness of 0.58 is introduced to account for the shielding effects of a 0.5-m breeding blanket that is not included in the ref. 15 analysis but is assumed to be present in all cases here. This factor is an estimate based on comparisons of more detailed shielding computations.

A.5 BUCKING CYLINDER

The bucking-cylinder stress analysis is also taken from ref. 15. First, the buckling pressure is required to be less than one-fifth of the critical buckling pressure of an open cylinder. The buckling pressure is the total TF coil inward centering force divided by the outer-cylinder surface area, and the critical pressure is

$$\sigma_p = 0.25 \times E_{bc} \times (t_{bc}/r_{bc})^3, \quad (A.3)$$

where

E_{bc} = Young's modulus of the bucking cylinder (2.09×10^{12} Pa) ,

t_{bc} = bucking-cylinder thickness,

r_{bc} = bucking-cylinder average radius.

Also, the bearing pressure is required to be less than two-thirds of the yield strength (414 MPa), where the bearing pressure = $2 \times$ buckling pressure/ $[1 - (r_{ibc}/r_{obc})^2]$, where r_{ibc} and r_{obc} are the inner and outer bucking-cylinder radii.

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