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AN OVERVIEW OF THE TPX NEUTRONICS AND SHIELDING ASPECTS

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ABSTRACT

The Tokamak Physics Experiment (TPX) device, which is under conceptual design in the US., presents several neutronics and shielding issues that are challenging from the design analysis viewpoint. This paper presents an overview of the various issues and a summary of the various design analyses that have been performed thus far to quantify the issues and to guide the design toward near-optimal engineering solutions to the problems. Some planned activities are also discussed.

I. INTRODUCTION

We have been involved in the neutronics and shielding analysis efforts in support of the TPX machine design since its inception. In this paper, we discuss the various design issues and the work that we have performed to guide the design efforts, in an approximate chronological order. The limited space here allows only very brief summaries. The interested readers are referred to the various references for details.

II. THE SEARCH FOR A LOW ACTIVATION MACHINE

In late 1991, many design concepts were being explored for the post-TFTR era. During this early stage, the only known constraint was that the machine should not cost more than about 400 million dollars. Most of the concepts centered around an advanced tokamak that would do advanced physics. Other technological features were also considered. One that was often emphasized was a low activation environment that would not require any remote maintenance equipment, thus saving some 20 million dollars.

In the kind of frenzied environment characterized by people offering a lot more questions than answers, we were interested in finding out how feasible this kind of claim was. Hence, we carried out simple 1-D survey activation calculations [1] for the Long Pulse Advanced Tokamak (LPAT) proposal, which assumed a lifetime neutron production level of about 6.5 x 10^{22} D-D neutrons. Table I shows the contact dose rates at the vacuum vessel at various cooling times, assuming the entire machine was made with each of the various materials.

These results showed that while there are pure elements such as aluminum and vanadium which, in a pure D-D neutron environment (without D-T burn), will have very low residual activity levels, the presence of small concentrations of impurities or a small fraction of D-T neutrons may lead to more than a million-fold increase in the activation levels and disqualify them as low activation materials. Since there is no easy way of removing the tritons produced in D-D reactions and there is no adequate engineering database for constructing a tokamak out of sufficiently pure aluminum or vanadium, it was clearly premature to assume that this machine, or other proposed steady state machines, could be maintained fully 'hands-on'.

This simple study prompted people to be more realistic in making low activation claims. Subsequently, the neutron fluence level was reduced by about an order of magnitude and a few more survey calculations were performed on a larger combination of candidate VV and in-vessel materials that were thought to be interesting based on various other properties[2]. The results revealed that Cu was better as a divertor material than the other material combinations considered.

However, since the first wall tiles are typically brazed to the copper backing, the in-vessel dose rate would depend critically on what kind of braze material is used. Table II shows the results, using different braze materials and relatively benign Al and Ti alloys for the VV, after 6 x 10^{21} D-D neutrons [3].

We also performed comparative activation calculations, with various in-vessel materials, for the three contending options in early 1992, namely, 1) the superconducting coil option; 2) the resistive demountable copper coil option; 3) the resistive Al6063 wound coil option [4].

Much like Tables I & II, the results of these calculations again revealed that one would be hard pressed to reduce the in-vessel dose rate to below 100 mrem/h about a week after shutdown, a level which begins to look feasible for limited 'hands-on' maintenance, *while satisfying all the other engineering constraints.* Vanadium was ruled out primarily because of the high development cost required. Aluminum was ruled out primarily because it would not withstand the bakeout temperature of about 350 ^{o}C required of the vacuum vessel. To get an activation level that is low compared to steels and inconels, which would be the favored materials from other standpoints, Ti6Al4V turned out to be a good compromise. Even though it is not strictly a low activation material that will eliminate the need for remote maintenance, it nevertheless is much better than steels and inconels in that respect. Moreover, this widely used material has good mechanical, electrical, manufacturing, vacuum properties and reasonable costs. Hence, it was chosen as the preferred material in structural applications.

Instead of looking for a design that would not require remote maintenance, it was evidently much more feasible to control the in-vessel materials and the initial neutron production to allow 'hands-on' maintenance during the first one or two years of operation. Calculations indicated that, with copper and braze materials present, the contact dose rate would be < ~10 mrem/h one week after shutdown, if the neutron production could be limited to < ~2 x 10^{20} D-D neutrons [5].

III. NUCLEAR HEATING MATTERS TOO

By February 1992, the superconducting option (SSAT-S) had emerged as the favored option. With this choice, the issue of nuclear heating in the superconducting coils quickly became a primary concern. To limit the nuclear heating rate, shielding would need to be installed between the vacuum vessel and the coils. It would therefore be advantageous to design the shield such that the delayed dose rate levels in the ex-shield regions be low enough for mostly hands-on maintenance. Simple 1-D survey calculations of the nuclear heating rates [6, 7] and of the contact dose rates [8] were performed. Assuming perfect shielding, it would take about 10 cm of borated water to reduce the heating rate to about 4 kw at 3 x 10^{16} D-D n/s. It was obvious that these simple 1-D models would not be adequate in addressing the nuclear heating and ex-vessel activation issues because of the presence of huge penetrations such as the ports and the vacuum pumping ducts. Thus, work was begun on a 3-D model to address the issues.

IV. PRELIMINARY 3-D ANALYSES

We constructed a 3-D model using the MCNP [9] code by assuming that every bay was identical. To address the delayed dose rate problem, we employed the effective delayed gamma production cross section technique [10], and the multigroup transport option available in MCNP. Thus, we were able to examine both the prompt nuclear heating and the delayed dose rate effects using a self-consistent 3-D model that took into account the important 3-D transport effects such as the leakage through the major penetrations and the inter-coil spaces [11].

Subsequent calculations using the same technique took into account the presence of the neutral beam box [12]. These preliminary calculations demonstrated the power and usefulness of our computational approach in dealing with the complicated 3-D geometry.

The results showed that, with ~20-30 cm borated water shield outboard, the external environment could be brought down to a few mrem/h a week after shutdown, provided that the large penetrations were properly shielded. It was also necessary to have some shielding in the inboard inter-coil regions to prevent the PF-5 coils to become highly activated. Potential problem areas included the neutral beam ports where large openings would be required and there was very little space around the ports for shielding. Hence, shielding of the horizontal ports would have to be optimized individually because of the large variations in local geometry.

V. 1-D PARAMETRIC SHIELDING OPTIMIZATION

Even though the 3-D model gave us meaningful dose 'maps' based on the available information, it was too costly for systematic investigation of the shielding effectiveness and cost as functions of various parameters such as the D-T neutron fraction, the shield thickness and composition. Thus, a systematic parametric and optimization study was carried out in simple 1-D geometry and reported in Ref. [13].

This study looked at the cost effectiveness of various shielding combinations with respect to the following factors: 1) the choice of nuclear response as a measure of the shielding effectiveness; 2) the attenuation factor desired for the chosen measure; 3) the material composition of the shield and the unit cost of each constituent; 4) the distribution of the component materials in the shield; 5) the effective D-T neutron fraction in the D-D environment; 6) the cost of space for shielding.

Even though the low activation requirement ruled out many common shielding materials, there were still too many combinations that could be considered. We examined three representative combinations: 1) the low unit cost, hydrogenous liquid based "poor man's option" consisting of water, boron carbide and lead; 2) the intermediate unit cost, hydrogenous polymer based system consisting of paraffin, boron carbide and lead; 3) the strictly performance oriented "rich man's option" consisting of titanium hydride, boron carbide and tungsten.

It was found that a one-zone "poor man's option" made up of about 89% water + 1% boron carbide + 10% lead was very cost-effective under most circumstances. It would not be worthwhile to fine-tune the composition and distribution unless the shielding space cost much more than 2 million dollars per *centimeter*. Also, there was no incentive in pursuing the "rich man's option", unless the real estate cost significantly more than 1 million dollars per *centimeter*.

Following this analysis, M. Cole, *et al*, developed some conceptual hardware configurations and performed initial cost estimates [14]. It was concluded that a double-walled vacuum vessel containing water as the neutron shielding material with additional boron/lead tiles on the outside appeared to be more feasible and less costly than systems using titanium hydride or polymer based materials. One major constraint was that the 350 ^{0}C bakeout temperature would cause titanium hydride or polymer to decompose, while water could be drained and superheated steam be used in its place.

Subsequently, however, the mechanical design was carried out based on the results of 1-D analyses reported in Refs. [15,16], which showed a boron carbide layer of 1 cm followed by 2 cm of lead, with varying amount of pure water in the vacuum vessel. Since lead would not withstand the bakeout temperature, it was substituted with a lead monosilicate layer (essentially silicate glass matrix with lead oxide dispersed in it) of about 4 cm, containing the equivalent of about 1 cm of boron carbide loading. It was envisioned that the lead glass layer would be in the form of small tiles mounted on the external surface of the vacuum vessel with titanium studs. This concept was later developed by B. Nelson, *et al*, and presented as the baseline design in the TPX Conceptual Design Review.

VI. 2-D CALCULATIONS OF NUCLEAR HEATING DISTRIBUTIONS

The integral nuclear heating estimate given in Ref. [16] was judged to have an uncertainty factor of 2, which was felt by the TF design group to be too large for their design. To get a better handle on the TF nuclear heating rates, within the rather severe constraints on analysis resources, we attempted to approximate the 3-D geometry with a set of 2-D axi-symmetric models that examined the heating rates with various degrees of effectiveness in shielding the large ports [17].

Based on this set of calculations, taking into consideration the large uncertainties in the machine geometry, material distribution, and the limitation of the 2-D models, we recommended that a design level of 5 kw per 5 x 10^{16} D-D n/s (with 2% D-T) be adopted as the initial design level.

VII. ENVIRONMENTAL ASSESSMENT

The environmental aspects of the machine received very high priority and the assessment of the project was carried out in parallel to the pre-conceptual design. We performed calculations of the radioactivity inventories [18,19] which were then used the potential radiation doses at the site boundary and to the total population under normal and accident conditions. For the inventory in test cell air, the TFTR production rates per neutron [20] were used, since the machine geometry was still evolving, but the presence of shielding around the vessel would ensure that the TFTR numbers would represent the worst case scenario. Even with these conservative numbers, no significant impact has been found in the environmental assessment.

VIII. BASELINE TORUS SHIELDING OPTIMIZATION

After the Conceptual Design Review (CDR), the project got down to the business of refining the baseline design presented in the CDR. Concerns had been raised by some reviewers about the cost and reliability of having thousands of little lead glass tiles attached to the vacuum vessel and the mixed waste problem created by the use of lead in the tiles. To us, the issue was more fundamental. In our parametric study [13], we did not examine the multi-layered option because we found that the homogenized option, *with the same composition*, was always more effective. This led us to examine the cost effectiveness of the baseline design. A set of parametric calculations were carried out based on the water + boron carbide + lead glass combination [21,22]. The integral results were also verified with 2-D models.

The results showed that: 1) The effectiveness of boron loading quickly reached some 'saturation' level beyond which there would be very little gain, which meant that the CDR design was very wasteful of the costly boron carbide. 2) Homogenizing the materials improved the shielding effectiveness by at least 50%. 3) The most effective layered shield was less effective than the least effective homogenized shield using the same composition. 4) With sufficient boron in water (about 1% by volume of boron carbide equivalent), lead glass could be replaced with borated water, resulting in only < 5% increase in the nuclear heating rate compared to a homogenized shield with lead glass. This simplification would eliminate the tiles and reduce the cost. 5) Further cost reduction could be achieved by dissolving less expensive boric acid in water instead of using the expensive boron carbide to achieve the necessary 'neutron poisoning' effect.

We also examined the impacts on the delayed dose rates that might be brought about by these changes [24], using 1-D and 2-D models. The study revealed that the perturbations in the dose rates were appreciable primarily in the shield itself. Outside the shield region, the dose rates might increase somewhat, but they were of no significant practical consequences. Hence, we were able to conclude that the lead glass tiles specified in the baseline design served little purpose other than making the shield much more complicated and costly. It would be much less costly and more effective to use borated water containing cheap boric acid and eliminate the tiles altogether.

Independent 1-D calculations by R. A. Lillie [25] and R. T. Perry [26] confirmed the validity of our analyses and the reasonableness of our approach. Independently, the JT-60SU study also adopted the same shielding approach [27]. Hence, the borated water shield was adopted as the new baseline design. The shielding cost savings due to the design changes have been estimated at about 2 million dollars [28].

Tests are under way to examine the corrosion effects of borated water on titanium alloy. Preliminary data from some manufacturers and these tests indicate that it is not a problem.

The effectiveness of using the inboard space for additional shielding was also examined [29]. The results showed that about 30% reduction could be expected if ~12.4 cm of additional inboard water shield was installed. Thus, the inboard first wall coolant lines were rerouted into the double-walled vessel to make about 15 cm more space available for shielding and the TF coils, compared to the CDR design.

IX. 3-D CALCULATIONS OF NUCLEAR HEATING RATES

While we were busy optimizing the torus shielding, the TF design team had been examining the thermal behavior of the coils using the 2-D heating distributions generated previously [17], by assuming some distributions in the inter-coil structures and certain toroidal 'peaking factors' in the winding pack. The results indicated that the critical temperature margin was only about 0.1 ^{o}K. As such, it was proposed to increase the size of the coils to increase the design margin. In the Critical Configuration Review held in late January ,1994, we cautioned that it was premature to draw such conclusions from the 'fudged' 3-D heating distributions which might be overly conservative.

Hence, we set out to calculate the 3-D heating distributions with a 3-D model that represented a 1/64 section of the machine [30]. With this model, we were able to calculate the 3-D heating distributions for D-D [30] and D-T [31] operations, and for various shielding options, to within the spatial resolutions specified by the TF design group. The integral heating rates in the winding pack turned out to be about 14% less than the corresponding 2-D results [14], confirming our expectation that they were somewhat conservative, as intended.

Estimates of the total heating rates in outer PF coils have also been obtained based on the 3-D model [32]. As of this writing, these values are being used to evaluate the coil performance.

X. CONCLUSIONS AND FUTURE PLANS

Many TPX neutronics and shielding issues of importance have been analyzed at various levels of details to support the conceptual design. More analyses at higher levels of details are planned. Examples are: 1) optimization of the shielding around the vacuum pumping ducts; 2) shielding design for the LH/ICRH launchers; 3) activation dose rate distribution calculations based on updated machine geometry; 4) shielding design and analysis for the other major penetrations; 5) shielding design and analysis for the diagnostics.

Most of these will require multi-dimensional models for accurate analysis because of the complex geometry. However, with the advent of powerful low-cost desk/lap/palm-top computers, computational efficiency is no longer the major bottle-neck for the routine solutions of these challenging multi-dimensional problems. Rather, it is the poor user-interfaces of the various computational tools that are making life in the multi-dimensional world so much more unpleasant. Good friendly user-interfaces are obviously long overdue.

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Table I: Contact dose rates (mrem/h) at the VV for three cooling times and various materials.

	Pure D-D Neutron Spectrum			D-D + 1% D-T Neutron Spectrum		
	1 week	1 month	1 year	1 week	1 month	1 year
Inconel 625	1×10^5	8×10^4	1×10^4	1×10^5	8×10^4	1×10^4
Al*	$< 10^{-7}$	$< 10^{-7}$	$< 10^{-7}$	18	4×10^{-2}	4×10^{-3}
Al6061	50	40	13	80	45	16
Steel	2×10^5	1.5×10^5	1.6×10^4	2×10^5	2×10^5	1.8×10^4
V*	$< 10^{-7}$	$< 10^{-7}$	$< 10^{-7}$	1×10^3	0.4	$< 10^{-7}$
Ti*	1.5×10^3	13	$< 10^{-7}$	5×10^3	2×10^3	1×10^2
Cr*	1.5×10^4	1×10^4	2.0	2×10^4	1×10^4	2.0
V15Cr5Ti	2×10^3	1.1×10^3	1.3	3×10^3	1.4×10^3	7.0
Ti6Al4V	1.5×10^3	13	$< 10^{-7}$	5×10^3	2×10^3	1×10^2

* assumed 100% pure.

Table II: In-vessel contact dose rates (mrem/h) at two cooling times for various divertor/first-wall/VV material combinations.

VV Composition	Divertor/First-wall Composition	1 week	1 month
Ti alloy	90% C + 10% Cu	110	60
Ti alloy	89.5 % C + 0.5% Braze 1 + 10% Cu*	150	80
Ti alloy	89.5 % C + 0.5% Braze 2 + 10% Cu*	500	500
Ti alloy	89.5 % C + 0.5% Braze 1 + 10% Mo alloy*	600	70
Al alloy	90% Be + 10% Al alloy	30	25

* Braze 1: 70% Ti + 15 % Cu + 15% Ni; Braze 2: 68.6% Ag + 26.7 % Cu + 4.5% Ti

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