Development of a Transient Coaxial Helicity Injection for Solenoid-free Plasma Start-up and Subsequent Non-inductive Sustainment

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1. Technology to be assessed:

Transient Coaxial Helicity Injection (CHI), for plasma start-up without reliance on the central solenoid.

2. Application of the technology

As shown in Figure 1, CHI is implemented by driving current along field lines that connect the inner and outer lower divertor plates. The standard operating condition for CHI in NSTX uses the inner vessel and lower inner divertor plates as the cathode while the outer divertor plates and vessel are the anode. A CHI discharge is initiated by first energizing the toroidal field coils and the lower divertor coils to produce magnetic flux linking the lower inner and outer divertor plates, which are electrically isolated by a toroidal insulator in the vacuum vessel. After a programmed amount of gas is injected into the vacuum chamber, a voltage is applied between these plates, which ionizes the gas and produces current flowing along magnetic field lines connecting the plates. The method initially drives current on open field lines. During transient CHI, by appropriately shaping the injector flux, and after the injected plasma fills the vessel (Fig. 1), by rapidly turning off the injected current, the injected open flux could be made to close in on itself to generate a closed flux configuration, as shown in Figure 1.
The concept is applicable to tokamaks and STs. CHI would generate a large fraction of the initial plasma current by injecting open poloidal flux into the vacuum vessel. Through a process of magnetic reconnection, the injected flux closes in on itself to generate a closed flux equilibrium. Though the method seems unconventional, it works as demonstrated by these major results: Initial demonstration on HIT-II and on the much larger NSTX devices [1,2], demonstration of compatibility with standard inductive operations on HIT-II and NSTX [3,4], understanding the physics of magnetic reconnection leading to large volume flux closure [5,6].

3. **Critical variable(s)**

The critical variables are (1) the magnitude of the injected poloidal flux, (2) shaping of the injector flux, as this relates to the fraction of the flux that converts to closed flux, (3) physical limits on the maximum amount of poloidal flux that could be injected in a tokamak and ST that results in the generation of a stable closed flux configuration, and the requirements on the toroidal field, (4) transitioning from CHI produced plasma to steady-state non-inductively driven plasma configuration for steady-state sustainment, (5) needed improvements to the electrode configuration that allows the required insulator to be resistant to neutron damage for the life-time of the reactor.

At present about 20% of the total plasma current required for sustained operation has been generated by transient CHI. The present understanding suggests that it may be possible to generate all of the needed current in a ST / tokamak using transient CHI. If this is the case, then one could transition directly from a CHI produced plasma to a non-inductively sustained plasma, without the difficult intermediate step that involves non-inductive current ramp-up.

Such a demonstration would enable a fundamental change to the tokamak / ST configuration that allows the central solenoid to be dispensed with. The new configuration would take advantage of evolving developments in high-temperature superconducting technology to develop a simpler design high toroidal field device that relies primarily on CHI for plasma current generation.

4. **Design variables**

**Injector flux:** The injector poloidal flux magnitude is controlled by the physical location of the divertor coils, and the currents driven in these coils. Typically three poloidal field coils are required to shape the injector flux to a configuration that leads to the appropriate reconnection in the injector region to generate closed flux.

**Toroidal Field (TF):** Present STs have operated with toroidal field values up to 0.55T. Increasing the magnitude of the toroidal field substantially to about 6 T, would allow the injection of large amounts of poloidal injector flux (~ 4 Wb) with modest levels of
injector current. With improvements to high temperature superconductor technology, a large ST with 1 meter of the inner core devoted to the TF center leg, and 50 cm neutron shielding, with $R > 3$ m, $A < 1.5$ should be possible.

Voltage requirements and power supply parameters: A short pulse (~10 ms) high-voltage (~10-20 kV) would be applied to electrodes inside the vessel to inject MA levels of current. Preset ST systems have operated at 4 kV and injected 40 kA level currents to generate high-quality plasma equilibrium. 10 kV, MA level currents have been injected in spheromak configurations, so these parameters have been reached successfully in present experimental devices.

PF coil locations: The location of the PF coils is important as it controls the injector flux and flux shaping. It is possible the divertor coils used for advanced divertor configurations (such as the super-x divertor) could be used to generate the injector flux [7]. At this location, the ceramic insulator is naturally shielded from neutrons. Experimental tests in these configurations are needed.

ECH heating: The electron temperature on CHI produced plasmas needs to increase to the >1keV using electron cyclotron resonance heating [8] so that current could be sustained using neutral beam current drive [9] or EBW current drive [8].

5. **Risks and uncertainties**

As indicated by the numerous published references, the method works at significant levels of generated current. So, there are no real risks. The work that remains is to determine how far the concept could be extended? What fraction of the current needed in a tokamak / ST could be generated by transient CHI? Could all the current needed for sustained operation be generated by CHI?

This requires both computational simulations as well as experimental testing in a number of present devices to establish any physics limits that may exist as the amount of injected flux is increased to very high levels.

The power supply and voltage needs have been demonstrated to a large extent in present devices, these would be simultaneously studied as injected flux level is increased in present devices.

6. **Maturity**

The method is currently at the TRL5 level of readiness. Supporting experiments in the following devices would bring it up to TRL6 level of readiness in 3 to 6 years.

QUEST ST in Japan: Test of new electrode configuration in support of reactor designs. This work is in progress. This is a medium sized ST test.
PEGASUS ST: Design and implementation of a high-injector flux configuration, with high toroidal field in support of studies in which CHI would generate currents at levels consistent with the existing PF coil set. This is a medium sized ST test.

NSTX-U ST: The injector flux capability in this device should allow current generation levels approaching a MA. The extensive diagnostics capability and the auxiliary current drive and heating systems capability would allow tests that would bring the concept to a TRL6 level of readiness, possibly to the TRL7 level of readiness. This is a large sized ST test.

DIII-D Tokamak: QUEST / PEGASUS-like electrode configuration test on DIII-D for a demonstration on a large tokamak.

7. Technology development for fusion applications

High temperature superconductor technology developments that allow the toroidal field to be increased above 6 T would be of considerable benefit to this concept, as this would make it easier to inject large amounts of flux using a smaller external power supply.

There has been steady good progress over the past 15 years. The main concept development itself primarily requires continued experimental testing on present STs (as noted above), and continued 3-D MHD simulations.

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References