

**Neutron Generator System
Design Report
for
Zurich University Neutron Facility
/CERN**

**for
Dr. Christian Regenfus, University Zurich, CERN**

CERN purchase order CA 1491984

**prepared by
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and
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1 Introduction

The research team led by Dr. C. Regenfus at CERN has stipulated a Design Report regarding the neutron generator with movable radiation shielding housing and neutron area dosimeter interlock system.

This document is a public release version of the delivered Design Report.

Its focus topics are as follows:

- Interlock safety features of the neutron generator control system
- Mechanical characteristics of the mobile shielding sub-system
- MCNP model analysis of the neutron generator induced radiation field within and external to the shielding.

2 NSD Neutron Generator Overview

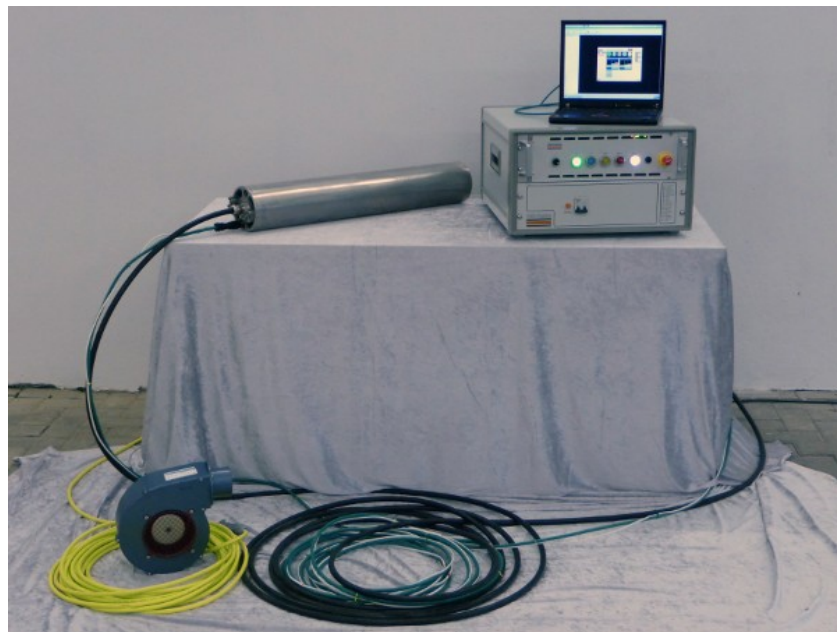
2.1 Neutron Generator Set

The NSD-Fusion Neutron Generator family includes a range of performance and configurations.

The NSD-NG-1E7-DD-C unit has a DC high voltage power supply rated at 120 kV, 15 mA maximum (1.8 kW).

This power is put into a glow discharge plasma load contained within the reaction chamber. The gas is Deuterium. This results in DD fusion and the emission of 2.5 MeV neutrons.

The neutron output is effectively regulated by an automation control computer. The user interface is via a LAN connection to a PC running a VNC client. Security functions are included.



NSD-Fusion Neutron Generator NSD-1E7-DD-C

2.2 IEC Fusion

The history of the NSD neutron generator technology is a branch of the history of Inertial Electrostatic Confinement (IEC) fusion research.

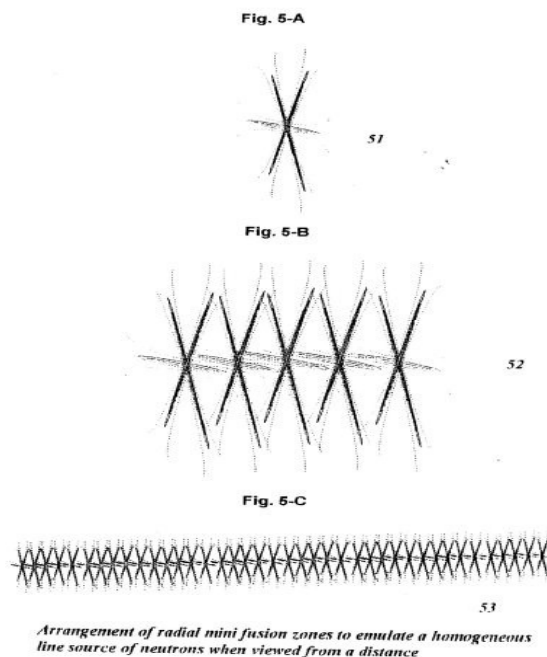
The inventor of television in the USA, Philo Farnsworth, also invented and named the IEC fusion concept (aka Fusor). In the late 1950's and 1960's he and a physicist R.L. Hirsch developed devices to attain higher fusion rates by the DD and the DT reactions. These were spherical geometry reaction chambers with a spherical central grid electrode.

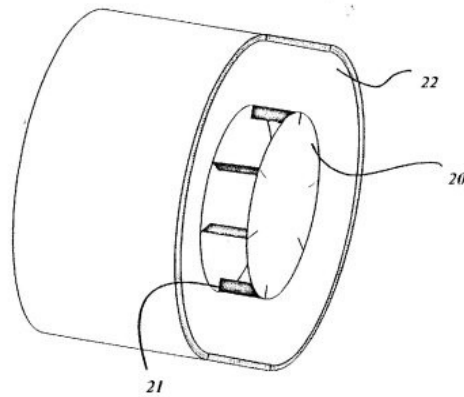
The concept is a simple glow discharge at low pressure of approximately 10^{-2} mbar. The DD or DT gas will break-down with the application of a voltage in the range 10 – 100 kV. In a steady state continuous power mode the plasma will be of a glow discharge type. Unlike classic parallel plate measurements and theory, the hollow cathode central electrode and outer anode topology defines a symmetric electrostatic field. The trajectory of a positive charge ion such as D^+ or T^+ or charged molecules of DD, TT, DT is such that it will generally be attracted toward the central cathode. The cathode is a cage-like structure with thin vane elements aligned radially. It is therefore apparently transparent in the local radial direction. Most ions pass through the cathode grid without colliding with it. Some ions born near to the cathode grid will collide with it. In the relatively high number density gas-plasma mix at 10^{-2} mbar a single ion will undergo charge exchange collisions with other ions and neutrals. The research literature indicates that this is an energy loss process that reduces the energy available for acceleration of ions to high kinetic energy where fusion grade collisions occur. It may never attain fusion power generation but it is fine for longest life industrial or research neutron generators.

The input power, operating temperature and waste heat transfer are significant engineering considerations. The thermal capacity or ability to radiate heat is an engineering limit. NSD will complete high temperature resistant electrode tests in order to determine the design limit for the cathode grid. Our design rule allows 1 kW or more per 25 mm of electrode length. High heat transfer with a water cooling loop allows the specific power to be increased.

The NSD experience with IEC devices has been to develop industrial grade automated robust neutron generators rather than experimental laboratory devices built from standard vacuum components and power supplies that are not well suited to the requirements of the device. NSD equipment is built for industrial specifications.

The diagram below illustrates the topology of the NSD reaction chamber fusion zone.





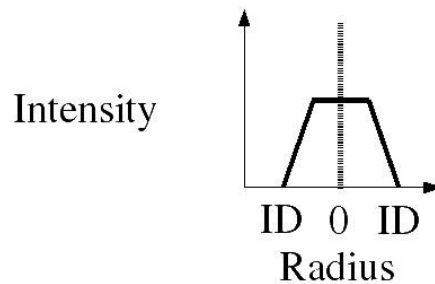
Schematic of a single cathode grid and anode cell

The compound alignment or stacking of electrode cells produces an almost homogeneous zone of fusion neutron emission along the length of the electrode. The star beams of electrons are aligned with the openings of the electrode. Electrostatic lens and ion kinematics draw the oscillating particles into the star beam channels. The electron beams carry a significant fraction of the power to the vessel wall. The electrode is a cathode and becomes hot enough for thermionic emission of electrons. The electrical energy becomes parasitic heat. The absence of a solid target spot and electron beam spots avoids very intense localized heating.

2.2.1 Fusion Zone

The fusion zone, where neutron emission reactions occur, is postulated to be as described below.

Measurements of the spherical Mk0 reactor chamber indicate that the neutrons are emitted with an intensity distribution as shown below.



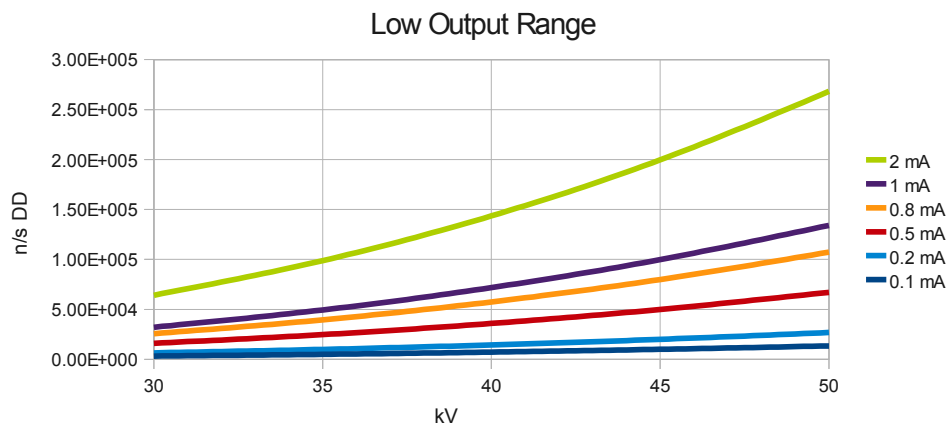
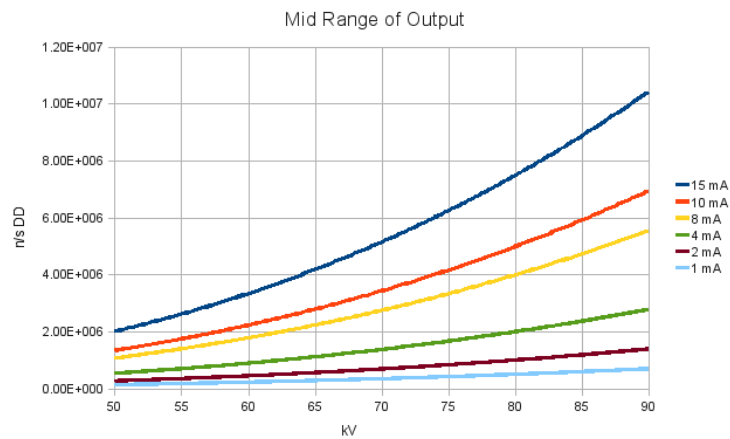
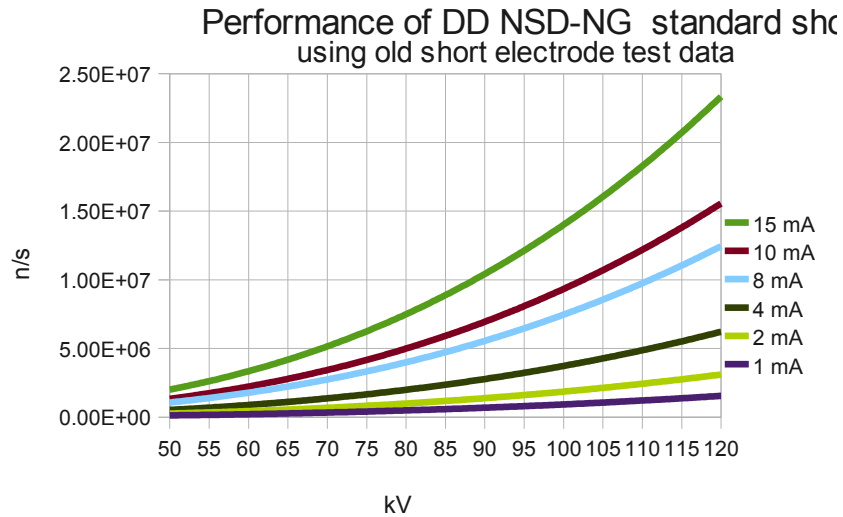
The intensity distribution within the linear reaction chamber is assumed to be similar. The plateau may correspond with the diameter of the central electrode. It may extend beyond the volume of the electrode. The plateau may be evidence that the formation of potential wells, often postulated for Inertial Electrostatic Confinement fusion, is not occurring in the operating regime of relatively high gas pressure. The flatness of the plateau has not been quantified.

The increase in intensity towards the centre is driven by the volumetric concentration of energetic ions and neutrals. There are many plasma mechanisms which may be considered. Most likely the ramp is not linear.

A limiting value of fusion rate may be zero at the inside wall of the chamber. However there may be some DD fusions between particles in motion and static particles in the metal matrix of the wall.

2.2.2 Neutron Yield

The blue line in the chart below shows the performance of the deliverable neutron generator model NSD-NG-1E7-DD-C. This unit has a 36 mm long electrode which is matched to the maximum input power of 1.8 kW.



2.2.3 Sealed Reaction Chamber

The reaction chamber assembly consists of a hermetically sealed vacuum vessel.



The Ultra High Vacuum seals are used in many vacuum systems including particle accelerators. Residual Helium leakage is less than 10^{-10} mbar.l/sec . At such a leakage rate the reaction chamber will retain operational conditions for a decade or longer.

In the event that reaction chamber servicing is required, the fill port can be opened. It is reusable. Servicing is offered by NSD-Fusion.

The entire reaction chamber is housed within an aluminium cylinder which has a cowling to direct the cooling air flow across the cooling fins.

2.2.4 Temperature Control

The "source" is the plasma with fusion grade collisions. It has a temperature of perhaps 30°C when operating.

The electrons emitted by the ca. -100 kV electrode cause heating of the vessel wall (and Bremstrahlung x-rays). The reaction chamber wall is also a heat exchanger (via the cooling fins). An equilibrium is achieved with a moderate air flow so that the base of the fins have a temperature of about 90°C at maximum input power. The outer casing or cowling has a temperature of about 30°C while the air is flowing.

The hot electrode will have a temperature of about 1000°C.

The hot getter pump has a temperature of 500°C. It is insulated by the vacuum and by an internal thermal cover. So it only needs 8 W to remain at operating temperature. When getter pump heater is switched off, it takes more than 30 minutes to cool down to ca. 100°C.

A concern sometimes expressed is that the hot running neutron generator might cause dangerous overheating of surrounding neutron shielding material such as paraffin. The following analysis shows that this is not possible.

IF the full powered NSD neutron generator suffers a cooling fan failure (no cooling air) the temperature will rise. A limit value for the cooling fin root temperature will be exceeded and the automated control will stop the high voltage and issue a warning or explanation message.

A simple calculation may be as follows.

Assume 1 kg of Aluminium. At 120° C versus 20°C this has a thermal energy increase of 8.97E4 J (specific heat of Al 24.2 J/mol.K)

Assume a cylinder of Paraffin 1 cm thick in contact with the Al body 0.3 m x 1.3 m diameter => 1.23E-3 cub.m of wax

For a 30 °C rise from 20 to 50°C melting point 7.31E4 J is required.

So the heat from the hot reaction chamber (not being cooled by flowing air) could in this very simple model transfer to the adjacent wax and heat it to above melting point.

However, the reality of heat transfer is that there are:

- insulation effects,
- other parts absorbing the heat and conducting it away
- general dilution by all of the mass not considered above

This means that it will be very difficult to locally heat Paraffin or similar shielding material to the point where it would melt.

Reaction chamber cooling is achieved by a variable speed centrifugal air blower. The 230 Volt DC motor rotation speed is controlled via a 0-10 volt control signal from a PID control loop. Therefore the target reaction chamber wall temperature is maintained. When the neutron generator is operated at lower than maximum output the air blower speed will also be reduced. This will mitigate air blower noise.

2.2.5 Tritium

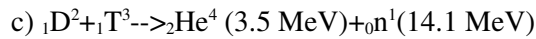
The Deuterium Deuterium fusion reaction will produce Tritium and Helium 3 nuclei at the same rate as neutron production.

The following fusion reactions predominate.

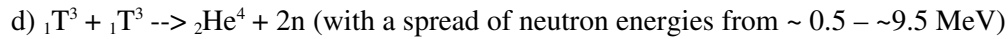
- a) ${}_1\text{D}^2 + {}_1\text{D}^2 \rightarrow {}_2\text{He}^3 (0.82 \text{ MeV}) + {}_0\text{n}^1 (2.45 \text{ MeV})$
- b) ${}_1\text{D}^2 + {}_1\text{D}^2 \rightarrow {}_1\text{T}^3 (1.01 \text{ MeV}) + {}_1\text{p}^1 (3.02 \text{ MeV})$

reactions a) and b) occur with equal probability.

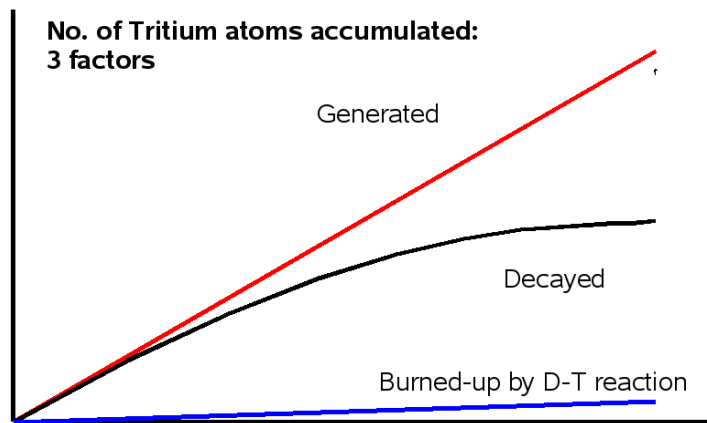
OR in the case of DT mix



OR in the case of Tritium only mix



In a DD neutron generator such as the unit to be delivered under CERN purchase order number CA 1491984, the accumulated or generated T will remain insignificant for the generation of 14 MeV neutrons from a 2.5 MeV DD neutron generator.



The above chart shows the Tritium accumulation factors. Decay is significant but burn-up is more effective in elimination of the accumulated Tritium.

So the produced and actually accumulated amount of Tritium remains well within the allowed limit of 400 GBq. A DT 14 MeV version of the NSD-NG will be loaded with Tritium to near to the 400 GBq limit. The DD 2.5 MeV version of the NSD-NG which is considered here may generate perhaps a few GBq of Tritium that will exist for any long period.

During early testing and the initial running in phase, the low pressure is established by means of a vacuum pump unit which exhausts to atmosphere. The emission of T from the vacuum pump unit at the possible rate of production remains below permitted levels.

2.2.6 Transport Container Classification

Two scenarios are considered:

- A Deuterium filled reactor chamber assembly will accumulate Tritium through its use as a neutron generator.
- A Deuterium-Tritium filled reactor chamber will lose Tritium through its use as a neutron generator.

The greatest amount of Tritium occurs in the DT version at the time of filling.

The case of the Deuterium filled reactor chamber will not achieve the same level of Tritium activity after a maximum practical working lifetime.

Therefore a classification for the case with the greatest amount of Tritium will cover all other scenarios of usage and transportation.

During transportation the Tritium is stored in solid solution within a getter pump material. The Tritium is incorporated into the instrument. The instrument remains hermetically sealed until it is returned for removal and replenishment of the reactant gas. The containment vessel is shown to be of very robust construction which can easily assure safety during minor mishaps and many accident conditions.

The allowable value of 400 GBq is based on the IAEA Regulations for the Safe Transport of Radioactive Materials 1996 Edition (Revised) No. TS-R-1 (ST-1, Revised)

Therefore the classification as a transportation package is:

UN2911 Class 7

This also satisfies the USA NRC Regulations

Title 49 CFR Part 173.421 for excepted radioactive material.

2.3 Central Control Unit

The automation platform is an Industrial PC with appropriate input/output modules for the neutron generator functions.

The DC High Voltage power supply is also accommodated in the same small rack enclosure as pictured.

The operator of the neutron generator first powers IPC unit on via the front switch.

After 30 seconds the software is boot loaded and operational. This is indicated by the green LED.

The control interface is a VNC client on another PC. Any common operating system is therefore supported after the open source VNC client application is downloaded and installed. A secure link is established by a log-in procedure so that only the operator's VNC client can be in interactive communication with the neutron generator controller VNC server.

Observers can also log onto the VNC server, but there will be no command capability. This may assist logging functions.

2.3.1 Operations Sequence

The operator selects the warm-up button on the virtual control panel. The getter pump heater temperature is gradually increased over about 30 minutes. During this phase the blue LED is at first constantly ON when the getter temperature is cold. As the heating progresses the Yellow LED flashes at an increasing rate to indicate that the heat-up is progressing. When the getter temperature is warm the blue LED goes OFF. When the Getter Heater is hot the Yellow LED remains continuously ON. After a predetermined waiting period which ensures that the gas pressure has stabilised, the neutron generator reaction chamber is ready to be operated.

Operation of the high voltage power supply is only permitted when the door and emergency stop interlock circuits are closed.



NSD Central Control Unit and 1.8 kW DC HV PSU enclosure. A typical industrial PC and its I/O modules plus power switching relays and 24 volt DC power supply can be accommodated in a 3U 19" sub-rack or similar enclosure.



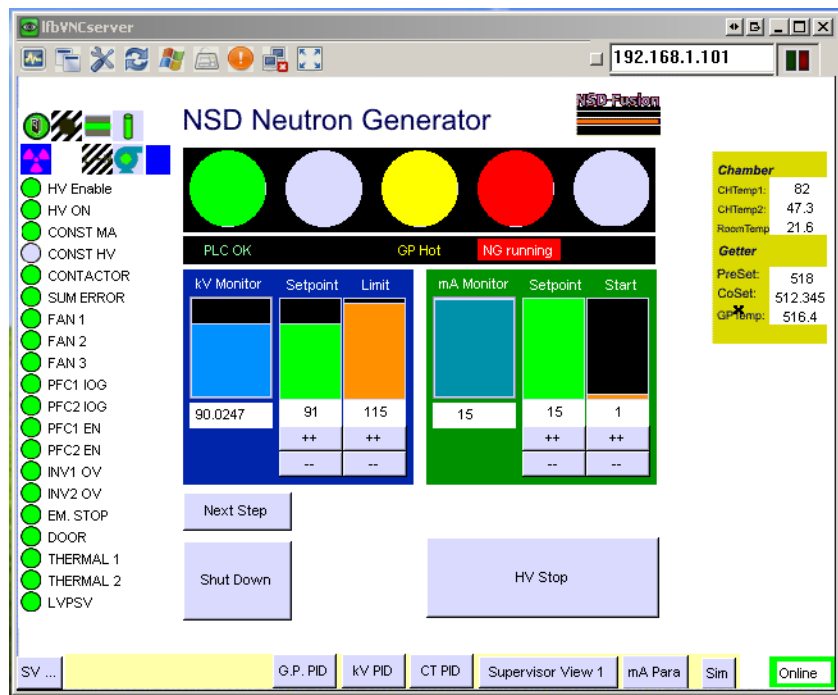
Connections: Data, Temperature, Heater Power, Interlocks, Warnings, HV PSU monitor and control, relay switched mains power out, mains power in.

The HV power supply requires the High signal (24 volt) directly to close the internal relay.

The automation also requires the High signal from the door and emergency stop circuits before it will issue the two high voltage enable signals which are also required by the high voltage power supply.

The High Voltage can be stopped manually by pressing the HV Stop button or the Shut-down button on the virtual control panel. Pressing a physical Emergency Stop button also immediately inhibits the High Voltage and hence radiation emission.

A second press of the Shut-down button will prompt an Accept or Cancel message. Acceptance will cause the getter pump heater to be switched off. After this the CCU will continue to monitor the reaction chamber temperature and operate the cooling air blower if necessary until the automation computer is switched off at the front panel main switch.



HMI control panel. Simpler control panel layouts can be provided.

The absence of front panel control switches on the HV PSU and CCU prevents independent command inputs.

A 15 m high voltage cable is provided so this allows the CCU and PSU to be located in another room if required.

The location of the control operator can be anywhere with Local Area Network connection. Internet external world connection is NOT recommended as it may be a security risk.

It is up to the operators of the neutron generator to determine the safety practice and configuration in accordance with ALARA criteria. The NSD neutron generator system provides flexibility.

Please refer to the user manual for more details about the monitor and command interface as well as the functions of the automation software.

The external interlock circuits connect to J3 on the back panel of the CCU.

DSWP	DoorIntl
EmStP	EmStop
VentEN	Dosi1
Caution	Dosi 2
Acoustic	Dosi 3
Operate	Dosi 4
Alarm	0V
0 V	0V

J3 connections

Note that the Neutron Dosimeter lines are available for a LB111 area dosimeter.

The front panel main power switch on the HV PSU must be in the ON position but the HV PSU will not become active because of the still open mains power relay in the CCU.

Then the mains power relay to the HV PSU is closed when the software is satisfied that all door and emergency stop interlock signals are HI. Then the HV PSU comes online.

Simultaneously relays will close for auxiliary external indicator devices connected to J3 on the back panel of the CCU. These include the 24 volt signal for warning lights.

On the front panel and the virtual control panel the RED LED will initially flash when the HV PSU is powered via the mains relay in the CCU. When the HV is enabled the RED LED will be continuous.

An audio alarm unit will also emit sound to warn that the HV PSU has been activated. It will again warn when high voltage is enabled.

A White LED provides an indication when the Emergency Stop or Door switch is opened.

The audio alarm will again emit a signal when such an event occurs.

Various fault monitoring functions in the HV PSU become active. All faults are indicated to the CCU software via dedicated lines in the umbilical cable.

If all safety conditions are satisfied, the software presents the operator with the virtual High Voltage enable switches. With both switches enabled High Voltage power will be delivered to the reaction chamber.

The software monitors the immediate plasma "ignition" and configures the system for normal regulation to achieve the pre-set target values of plasma voltage and current.

The plasma current is regulated by the HV PSU taking a target value from the CCU software.

The plasma voltage is regulated by the CCU software. This regulates Getter Pump temperature with determines the plasma voltage.

In the event of "mini-arcs" the CCU software determines if an unsafe condition is present. Mini-arcs are a phenomenon during the initial running in of the reaction chamber. With the high quality electrode used there is actually little electrode conditioning necessary.

The neutron generator will achieve the target operating conditions quite rapidly. As the control software evolves, the necessary "smarts" will be implemented to fully automate the rapid achievement of target operating parameters.

If the operator wants to switch off the neutron generation, the system will continue to maintain the standby conditions.

The utilization of the door and emergency stop interlocks is recommended for the laboratory safety protocol. The HV PSU shall be commanded to go off-line and the CCU mains power relay will isolate the HV PSU.

The opening of a door switch or an Emergency Stop switch will also ensure that this condition is achieved.

Similarly the CERN required Dosimeter interlock can be added. As there are four interlock or alarm signals provided by the Berthold Technologies LB111 neutron dosimeter these are used to provide warnings to the operator and to automatically trigger the CCU software to take the HV PSU off-line in the event of a safety limit being reached.

It is the function and responsibility of the CERN radiation safety officer to set the alarm and limit values and to place the neutron detection unit in a position which is meaningful for personnel radiation protection.

The CCU software will respond to warning and alarm signals. One Dosimeter interlock channel will be configured to initiate an Interlock LO signal that will take the HV PSU off-line.

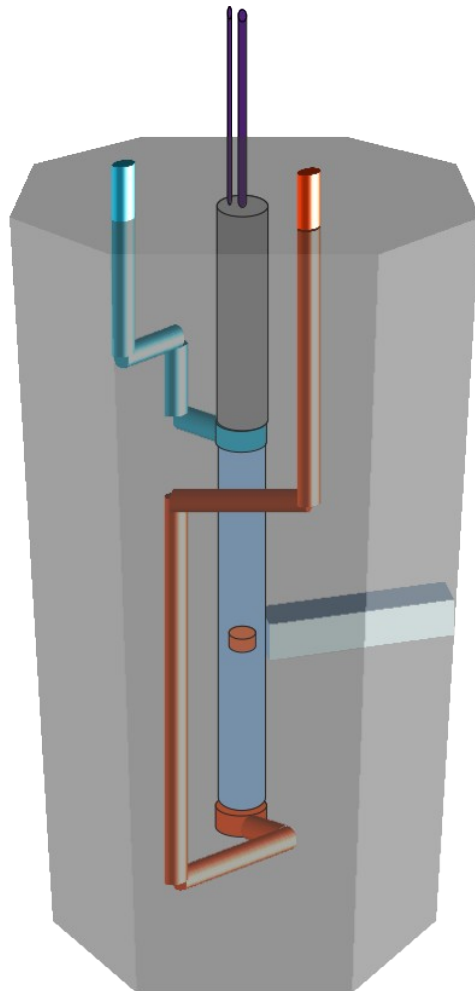
When operations are concluded, the operator can command the neutron generator to Shut Down. This will cut the Getter Pump heater power.

The air blower will continue to operate until the reaction chamber temperature is below a threshold value.

3 Shielding System

3.1 Concept

The diagram illustrates the basic concept.



Ambient cool air is blown into one duct. The flow is past the heat exchanger fins of the reaction chamber. The warm air exits the shielding via the other duct. Bends are provided to mitigate neutron streaming.

A hole or collimator is aligned with the neutron generator fusion zone.

A CERN requirement is high adjustment. The reaction chamber vertical position can be adjusted.

3.2 Mobility

A mobile or easily movable neutron and associated radiation shield has been included in the system specification.

A maximum diameter of 90 cm for the housing has been specified. This is mainly to limit the mass.

The height is to be limited to 2 m for doorway clearance. The width is limited to 1 m.

Access to the CERN laboratory is via double doors.

The pallet lifter option offers greater flexibility. The unit pictured here can lift 2500 kg.

The mobile shielding with the NSD neutron generator installed is shown below prior to its delivery to CERN.



Threaded rods are attached to the reaction chamber. These enable 40 cm of height adjustment of the reaction chamber within the shielding.

The High Voltage cable bend radius limitation must be respected.

3.3 MCNP Model

In order to provide optimal radiological protection capabilities, a Monte Carlo analysis was carried out using the radiation transport code MCNP5¹ (Monte Carlo N-Particle).

3.3.1 MCNP Model Geometry

A detailed geometry model was built reproducing the geometry and materials of the NSD-NG, and surrounded by a cylindrical shield of diameter 90cm and height 1.8m. The NSD-NG is located on the axis of the cylindrical shield, and can be moved to various positions along this axis within a bore which is slightly larger than the NG OD. A minimum of 30cm shielding material is provided at each end of the bore. It was assumed that, for the present application, the NSD-NG will be mounted vertically, with power cables entering approximately axially from above; appropriate cabling and feed-throughs are modelled in detail within the Monte Carlo study. Air ducting is required within the shield assembly to provide a throughput of sufficient cooling airflow to the NSD-NG (see section 3.1 above); this ducting occupies a small but non-negligible fraction of the shield cross-section, and therefore must also be represented appropriately in order to ensure that this does not result in a radiological weak point in the design. Four long steel rods be installed, equally spaced around the shield circumference, in order to provide a guarantee of structural integrity and to lock the assembly in place. These, together with a small air-gap around each rod, are also included in the model.

Additionally, a neutron beam exit port is located adjacent to the thin aluminium neutron “beam window” of the NSD-NG. At present, this exit port is the basic design: A square cross-section hole of side 10cm x 10cm, and positioned radially from the NSD-NG neutron beam window to the outside of the shield. The modular design of the NSD mobile shielding enclosure permits the retrofitting of enhance collimator modules, for example to optimise the delivered neutron spectrum characteristics, or to limit the output beam dimensions as required.

The detailed geometry of the NSD-NG internal engineering design is not discussed here. However, an extended source distribution reflecting the known NSD-NG source region geometry and characteristics is used for the radiological study.

3.3.2 MCNP Material Definitions

Note that, throughout, material definitions are specified using isotopic (rather than elemental) cross-sections. In each case, all naturally occurring isotopes of each element are including, in the appropriate natural relative abundances. This is highly desirable for modelling of neutron-gamma photon production processes.

¹ “MCNP – A General Monte Carlo N-Particle Transport Code”, Los Alamos National Laboratory

3.3.3 Candidate Materials for the Radiation Shield

An extensive range of materials has been considered in order to optimally meet the requirements of the current application. First and foremost, it is necessary to moderate the fast neutrons emitted from the NSD-NG, and then capture the resulting thermalised neutrons. Since we are dealing here with neutrons from dd fusion (2.45MeV), we will best achieve moderation using light elastic scatterers. Therefore, the most effective materials will be hydrogen rich solids; these also offer strong thermal neutron absorption via hydrogen capture – a topic which will be revisited shortly.

Traditional neutron shielding materials, including paraffin, while providing good moderation and absorption of neutrons, have less than optimal mechanical and structural properties, and in the present application, it is desirable that the entire structure be physically robust. A number of materials, including plastics such as polyethylene (PE), were considered. However, a particularly appealing product which is now being used for neutron shielding is Water Extended Polyester, known as “WEP”. This offers good neutron moderation and absorption due to its hydrogen content, but also permits moulding into appropriate shapes, curing to yield a strong solid. Another highly attractive property is the ease with which additives can be blended into the WEP mixture, for example if required to enhance radiological transport properties (see below). After comparison with other candidates, it was decided that WEP would form the optimal base component of the shield.

Therefore, an initial simulation was performed for the NSD-NG neutron generator located within a cylindrical shield of pure WEP (density 1.08g/cc) of diameter 90cm and height 1.8m, with the NG source point assumed to be midway up the shielding cylinder axis. Note that, due to the internal construction of the NSD-NG, this does not result in exactly equivalent shielding at the top and bottom of the cylinder (in fact, a slightly higher degree of radiation shielding is provided in the upward direction due to ceramic components above the neutron source electrode region).

Advanced features of MCNP, including the mesh tally option, were exploited to generate a radiation flux and dose rate map over the curved surface of the cylindrical shield, as well as over each end of the shield. A range of variance reduction techniques were deployed in order to optimise statistical precision. Through this study, both neutron and photon fields are expressed, unless otherwise noted, as biological dose equivalent rates in Sv/h (or more usually, due to the low residual dose rates achieved, in $\mu\text{Sv/h}$) using the ANSI/ANS-6.1.1-1977 conversion factor parametrisation, and assuming a neutron source output rate of 2×10^7 n/s (DD) as called for by the present application. The term “dose rate” will be used interchangeably with “biological dose equivalent rate” unless otherwise specified.

Furthermore, note that the present discussion relates to photon dose rates arising from secondary gamma-rays generated by (n, γ) reactions – both neutron inelastic scattering and neutron capture. Photon dose rates from Bremsstrahlung within the neutron source electrode are subject to additional dependences and are discussed separately (below).

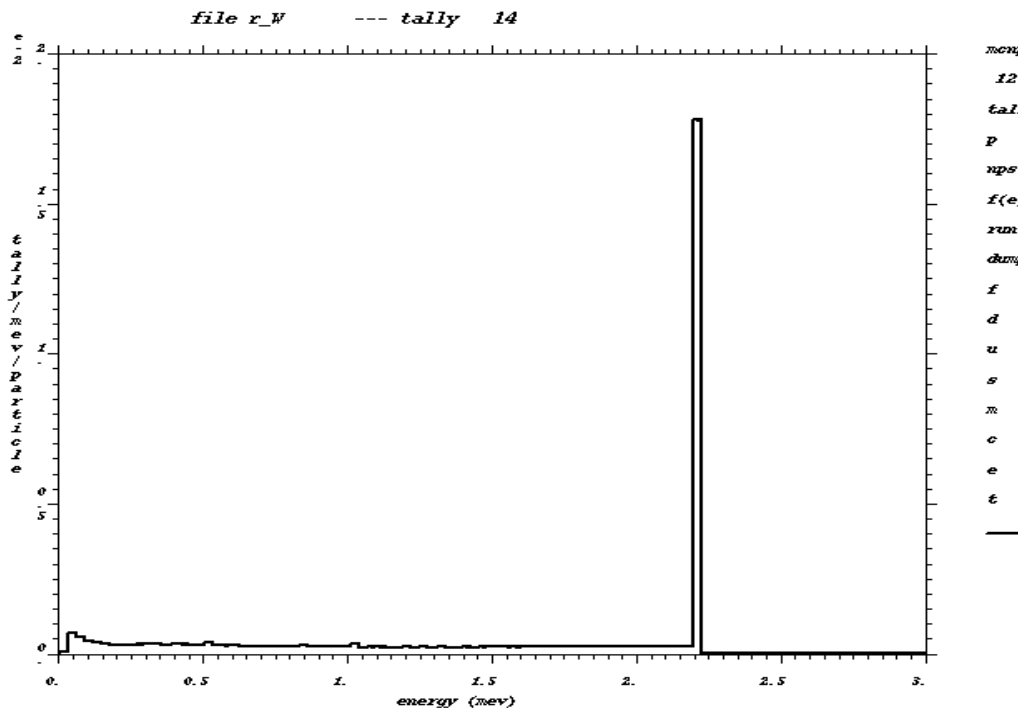
The initial models showed a maximum field for both neutrons and photons around the circumference of the shield in the plane of the neutron source region and collimator. The fields decreased for axial positions closer to the ends of the shield, and the top and bottom of the shield exhibited significantly lower fields for every case studied. Therefore, throughout this report, with the emphasis on radiological protection, the “worst case” fields close to the

source point are quoted. These are averaged over a band around the cylindrical surface of the shield, of width $\pm 20\text{cm}$ either side of the centre of the neutron source electrode region and collimator exit, but excluding the neutron beam exit hole and immediate surround. Dose rates around the circumference of the cylinder (at a fixed height) vary moderately, which is ascribed to the presence of air ducts and other non-axially symmetric structures; however, this variation in rate is small compared to the average value of the dose rate.

The initial model, with shield consisting only of WEP (density 1.08g/cc), provided an estimated maximum neutron dose rate of $3.03\mu\text{Sv/h}$, and an estimated maximum photon dose rate of $7.17\mu\text{Sv/h}$.

Thus, of a total (maximum) dose rate at the surface of the shield of $10.2\mu\text{Sv/h}$, over 70% of the dose arises from photons (rather than neutrons). We note also an earlier comment from CERN that “there is a considerable amount of gamma radiation produced”. As a result, additional optimisation effort was deployed in an attempt to reduce the surface photon field still further.

First, it is important to consider the mechanisms giving rise to this γ -ray field. The simulated γ -ray energy spectrum for a small test region close to the shield at the region of maximum field is shown below for the case of a shield comprising only WEP:



As can be seen unambiguously, the photon spectrum at the surface of the shield is overwhelmingly dominated by the 2.22MeV hydrogen (thermal neutron) capture line at 2.22MeV, together with background arising from scattering of this line. (Note that the plot shows the surface flux spectrum *converted to dose equivalent rate* and therefore does not show precisely the same shape as a γ -ray energy spectrum as measured by a spectroscopy detector; furthermore, structures below full-energy normally seen in measured spectra due to scattered photons escaping the detector are also absent; and finally, no detector resolution effects are folded into this modelled dose spectrum).

Note that, because of the relatively low neutron energy produced in dd fusion, there is virtually no inelastic photon production. Other than a trace presence of N and Sn (at about 0.1%wt levels or below), WEP consists of C, O and H. For obvious reasons, H does not give rise to inelastic scattering γ -rays, and carbon and oxygen have inelastic scattering thresholds above the dd neutron energy due to their high-lying first excited states. Therefore, the neutron-induced photon spectrum (i.e. the photon spectrum other than from source Bremsstrahlung) arises predominantly from thermal neutron capture for a dd source.

This important aspect is also reflected in another feature of the simulated radiation fields: Analysis of the neutron dose rate spectrum for the WEP shielding shows that only 1.08% of the surface dose, as measured in the region of maximum field on the curved surface of the shield close to the NSD-NG source electrode region, arises from thermal neutrons, while almost 99% arises from fast neutrons. (For the purposes of the present study, the epithermal neutron regime – typically considered to be the energy range 0.2eV to 10eV – is included in the category of “thermal” neutrons, and fast neutrons are taken to be those of energy 10eV and above.) Thus we see that, unsurprisingly, most neutrons which are moderated to thermal (or epithermal) energies are then captured within the shield, giving rise to capture γ -ray emission.

In order to reduce the photon radiation field at the surface of the shield, therefore, we have two options: (a) increase shielding to further suppress these γ -rays, or (b) reduce the rate of thermal capture on hydrogen.

Considering the first option (a): Due to the high energy of the hydrogen capture line, we note that interaction of these 2.2MeV photons with matter in the shield will be predominantly via Compton scattering. The Compton scattering cross-section varies approximately with the atomic number of the scattering nucleus, Z , and shielding efficiency increases roughly with the mass density of the shielding material. Thus it will be helpful to increase the shield density. However, we are heavily constrained in this regard, since it would be counter-productive to reduce the hydrogenous content of the shield, leading to an increase in neutrons reaching the outer part of the shield – potentially increasing both neutron *and* photon fields at the surface of the shield. However, it is possible to consider either small amounts of additives, or somewhat greater fractions of materials with significant hydrogen content; these options may offer a modest improvement in γ -ray shielding efficacy.

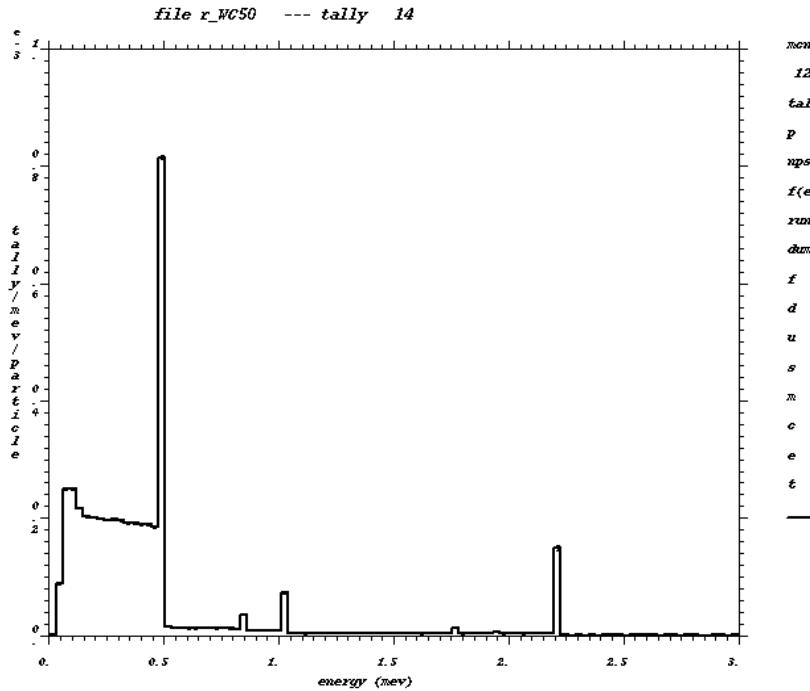
Option (b) offers alternative and more hopeful prospects: By reducing the rate of thermal capture on hydrogen, we reduce the rate of production of these highly penetrating 2.22MeV capture γ -rays correspondingly. However, we wish to reduce neither the neutron output from the NSD-NG, nor the hydrogen content of the shield, since this is required to achieve moderation of the initial fast neutron flux. How then can we reduce the rate at which thermal neutrons are captured by hydrogen? Instead, we propose adding boron-containing additives to the shielding material. Boron-10, which comprised 20% of naturally occurring boron, is an extremely effective absorber of thermal neutrons; indeed, the macroscopic thermal absorption cross-section Σ (the atomic cross-section multiplied by the number density of atoms per unit volume) for boron is more than 200 times greater than that of hydrogen. The effect of a small addition of boron is therefore to preferentially “mop up” thermal neutrons via capture on boron instead of via hydrogen capture (although some residual hydrogen capture will of course still take place). This not only means that the thermal flux is more rapidly suppressed, and therefore covers a smaller extent from the neutron generator, but crucially, capture on boron leads to the emission of a much lower energy capture γ -ray of energy 478keV (and

only with a 94% branching ratio) instead of the 2.22MeV γ -ray emitted by hydrogen. This in turn means that the resulting flux of capture γ -rays, which is unavoidable if the thermalised neutrons are to be absorbed, is now of very much lower energy, and therefore very much more strongly attenuated by the NSD-NG shielding material.

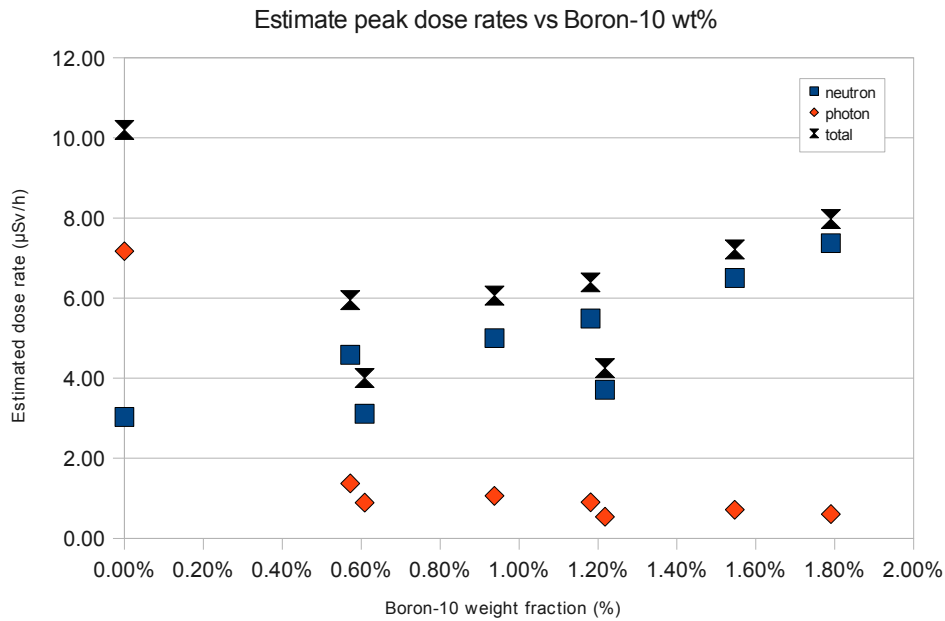
An acceptable way to add boron is via the addition of the naturally occurring mineral Colemanite ($\text{CaB}_3\text{O}_4(\text{OH})_3 \cdot \text{H}_2\text{O}$). Tests performed by NSD-Fusion show that this can be easily incorporated into the WEP mixture without compromising structural integrity. Additionally, Colemanite has a relatively high mass density (2.4g/cc), which will help to maintain the shield density and therefore assist with attenuation of γ -rays in the Compton scattering energy domain (~100 keV up to several MeV, typical of those produced by neutron inelastic scattering and capture). Finally, when source Bremsstrahlung radiation is discussed below, it will be noted that Colemanite also contains approximately 20%wt Calcium (atomic number $Z=20$), which will greatly enhance suppression of Bremsstrahlung photons (in the photoelectric “PE” energy domain below ~100keV, where absorption cross-section depends approximately on Z^4). Particularly desirable is the significant fraction of hydrogen within Colemanite – although only just over 2%wt, this is much lower than the hydrogen fraction in WEP (9%wt), but still preferable to the largely hydrogen free alternative sources of boron. For comparison, one such material, diboron trioxide (B_2O_3) will be considered below.

In addition to the pure WEP shield composition, therefore, MCNP models were also run for mixtures of WEP with 25%wt and 50%wt Colemanite, noting that higher Colemanite fractions are likely to compromise the structural integrity of the shield (as well as reducing the hydrogen content unacceptably), therefore limiting the amount of boron which can be added via Colemanite (which contains approximately 13%wt boron). In order to assess the benefits of further boron enrichment, therefore, five further cases were modelled, in which 10%wt of B_2O_3 (31%wt boron) was substituted for an equivalent mass fraction of either WEP or Colemanite.

The impact on the γ -ray field due to the addition of boron is striking. The figure below shows the surface dose spectrum, exactly comparable to the above figure for WEP and for the same neutron generator output, but in this case for a 50/50 mixture (by weight) of WEP/Colemanite:

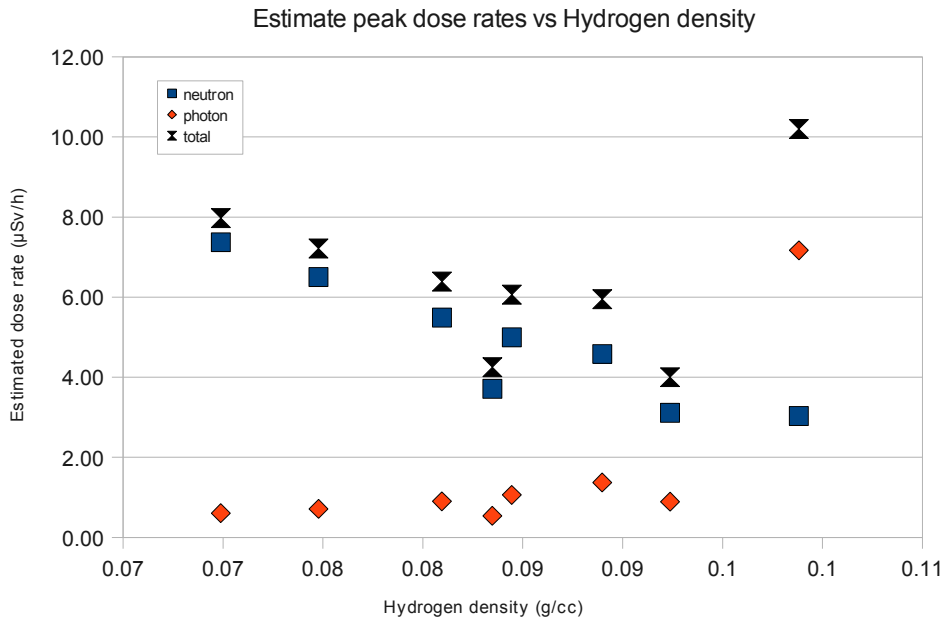
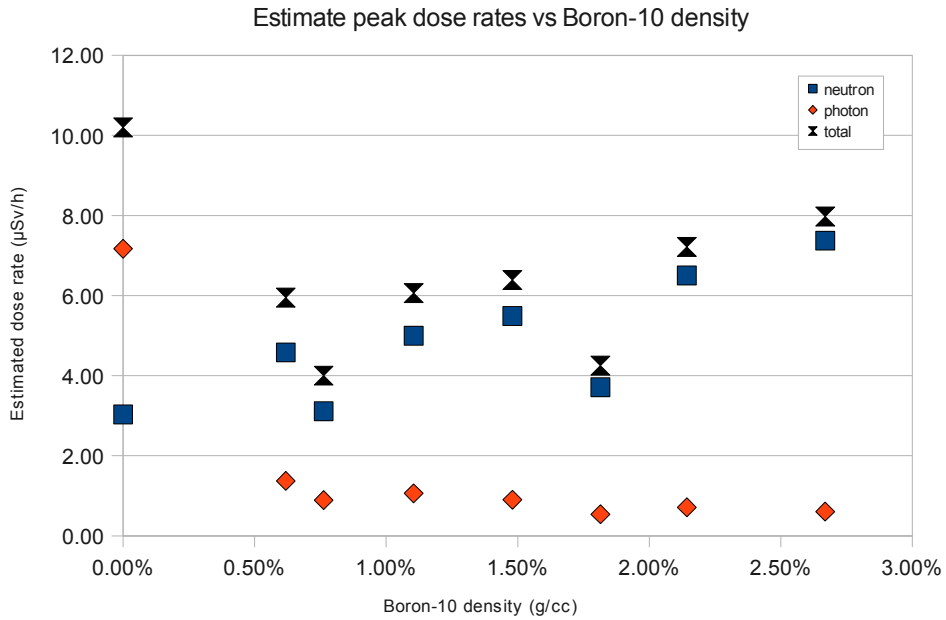


Note the change of scale on the y-axis – the maximum bin dose is now around 20 times lower than before, and is at 0.478keV instead of 2.22MeV as before. The hydrogen capture peak at 2.22MeV is still visible, but reduced in yield by approximately 2 orders of magnitude. The effect of replacing the highly penetrating 2.22MeV capture γ -ray yield by the much “softer” (and therefore much more strongly attenuated) 478keV line is also reflected in the calculated maximum photon dose rate at the surface of the shield, which drops from 7.17 μ Sv/h to only 0.54 μ Sv/h. Significantly, the photon flux now contributes only ~13% of the total surface radiation field, instead of over 70%, and thus the total radiation field at surface falls from 10.2 μ Sv/h to only 4.3 μ Sv/h.



At this point, it is worth remembering that, as pointed out earlier, the overwhelming part of the neutron dose rate at surface arises from fast neutrons. For pure WEP, this fraction is almost 99%, and this rises still further with any addition of thermal-neutron-suppressing boron. Therefore, we should not expect the addition of boron to substantially decrease the

neutron dose rate (since boron suppresses primarily thermal neutrons, which contribute only negligibly to the total neutron dose rate). However, the replacement of hydrogenous material by boron-rich (but non-hydrogenous) materials may well lead to less effective moderation of fast neutrons, and therefore a net increase in neutron dose rate.



3.3.4 Bremsstrahlung

As part of the radiological assessment, it is necessary to consider the photon dose rate arising from Bremsstrahlung in the NSD-NG high voltage electrode region. The Bremsstrahlung production rate depends on a number of factors which are not yet specified for the present deliverable system, and therefore it is not possible to simulate absolute dose rates due to Bremsstrahlung. However, it is possible to simulate the efficiency of the shielding enclosure as a means to suppress Bremsstrahlung radiation fields. This information can later be used, together with measurements of the Bremsstrahlung field from an unshielded NSD-NG, to deduce the residual photon dose rate arising. Two simulations have been performed: First, representing the Bremsstrahlung spectrum by an 80keV photon source, and secondly, as an absolute “worst case” to provide an upper limit to the Bremsstrahlung external dose rate, using a 120keV (line) photon source. In practice, the Bremsstrahlung spectrum will be a continuum from an end point at or below the NSD-NG HV energy down to effectively zero, and therefore will be suppressed more effectively than the line source model suggests. However, for a safety assessment, using a worst case scenario is appropriate.

The same geometrical models as above were used to estimate the shielding efficiency for Bremsstrahlung fields in the same peak dose region, around the circumference of the cylindrical shield close to the neutron source position.

Given the analysis for dose rate simulations, it seems clear that the addition of diboron trioxide has no benefit for the present application. Therefore, we focus our attention on combinations of WEP and Colemanite.

The following table shows the estimated external residual photon flux and dose arising from Bremsstrahlung relative to an unshielded NSD-NG (measuring at the same position outside the shield). Note that the suppression factors for flux and dose are not identical, since the shield alters the energy spectrum of the transmitted photons. As expected, the 120keV source is less well attenuated than the 80keV source. As before, these quantities are calculated to a precision of around 1%.

WEP+...	Assuming 80keV line photon source		Assuming 120keV line photon source	
	Residual relative flux	Residual relative dose rate	Residual relative flux	Residual relative dose rate
0%	2.52×10^{-2}	3.25×10^{-2}	5.96×10^{-2}	6.25×10^{-2}
25%	3.19×10^{-3}	3.48×10^{-3}	1.38×10^{-2}	1.29×10^{-2}
50%	3.83×10^{-4}	3.96×10^{-4}	3.00×10^{-3}	2.75×10^{-3}

It was noted earlier that a shield consisting of WEP with a Colemanite fraction in the range 25-50%wt would be optimal for minimisation of the neutron and photon fields arising from source neutrons, with a weak preference for somewhat lower Colemanite fractions (i.e. nearer to 25%) if neutron fields are of concern, although the effect is small. Clearly, if shielding the residual Bremsstrahlung field is particularly important, then a somewhat higher Colemanite fraction (of around 40%wt or higher) could be advantageous.

Note that, in the special case of Bremsstrahlung, very effective additional shielding techniques can be supplied. Relatively thin sheets of heavy metal (such as lead) can be wrapped around the NSD-NG body close to the source, effectively suppressing the Bremsstrahlung field. A hole would then be cut in the lead foil in front of the aluminium “neutron window” in the NSD-NG housing. Since the Bremsstrahlung is of energy below around 100keV, it will interact predominantly via the photoelectric effect (rather than by Compton scattering), and therefore will be confined to a “fan” close to the solid angle subtended by the neutron exit window. Therefore, further lead foil can be supplied on the outer surface of the WEP/Colemanite shield close to the neutron beam exit (or collimator). This will very effectively suppress any residual Bremsstrahlung photons.

3.3.5 Shielding Radiological Study Conclusions

A detailed Monte Carlo analysis has yielded the following conclusions:

- For suppression of the neutron dose rate external to the assembly, a shield consisting of WEP with an added Colemanite fraction in the range 25-50% offers excellent properties. If neutron dose rate is the sole criterion, then a Colemanite fraction towards the *lower* end of this range is marginally preferred.
- For suppression of the secondary photon dose rate resulting from neutron capture and scattering, the same shield composition is optimal. If photon dose rate is the sole criterion, then a Colemanite fraction towards the *higher* end of this range is marginally preferred.
- Taking the total radiological dose equivalent rate (including neutrons and photons), then a shield composition anywhere in the above range offers approximately similar shielding efficacy.
- If the residual field from Bremsstrahlung radiation produced in the NSD-NG is a concern, then a Colemanite fraction in the upper part of the above range would be effective in improving attenuation of the Bremsstrahlung flux. However, alternative targeted techniques for reducing this low-energy photon radiation may be preferable.

Note: The above Monte Carlo analysis is intended as guidance for design purposes only. The dose rate estimates provided should not be used for radiological purposes.

4 Shielding Performance Measurements

The next issue of this report will include a comprehensive survey.

5 Conclusion

The operational safety features of the NSD neutron generator take into account the interfaces required by radiation safety regulations and the considerations of ALARA (As Low As Reasonably Achievable).

The addition of a a shielding housing constrained to be mobile as defined above has been a new design exercise. The employment of MCNP as a design tool has been combined with a pragmatic cost minimization design study to achieve the customers requirements.

NSD-Fusion has manufactured an identical mobile shield for its own usage within its manufacturing and test facility.