nks
Nordisk kernesikkerhedsforskning
Norrænar kjarnöryggisrannsóknir
Pohjoismainen ydinturvallisuustutkimus
Nordisk kjernesikkerhetsforskning
Nordisk kärnsäkerhetsforskning
Nordic nuclear safety research

NKS-84
ISBN 87-7893-142-8

The Possibility and the Effects of a Steam Explosion in the BWR Lower Head on Recriticality of a BWR Core

B.R. Sehgal & T.N. Dinh
Sehgal Consult, Sweden

December 2002

Abstract

The report describes an analysis considering a BWR postulated severe accident scenario during which the late vessel automatic depressurization brings the water below the level of the bottom core plate. The subsequent lack of ECCS leads to core heat up during which the control rods melt and the melt deposits on the core plate. At that point of time in the scenario, the core fuel bundles are still intact and the Zircaloy clad oxidation is about to start. The objective of the study is to provide the conditions of reflood into the hot core due to the level swell or a slug delivered from the lower head as the control rod melt drops into the water. These conditions are employed in the neutronic analysis with the RECRIT code to determine if the core recriticality may be achieved.

Key words

Severe accidents, re-criticality, melt-coolant interaction

NKS-84
ISBN 87-7893-142-8

Pitney Bowes Management Services Denmark A/S, 2003

The report can be obtained from
NKS Secretariat
P.O. Box 30
DK – 4000 Roskilde, Denmark

Phone +45 4677 4045
Fax +45 4677 4046
www.nks.org
e-mail nks@catscience.dk

The Possibility and the Effects of a Steam Explosion in the BWR Lower Head on Recriticality of a BWR Core

Submitted to NKS and SKI

B.R. Sehgal

T.N. Dinh

Sehgal Konsult
Lill Jans Plan 4
114 25 Stockholm, Sweden

Table of Contents

Executive Summary

The analysis performed in this study considered a BWR postulated severe accident scenario during which the late vessel automatic depressurization brings the water below the level of the bottom core plate. The subsequent lack of ECCS leads to core heat up during which the control rods melt and the melt deposits on the core plate. At that point of time in the scenario, the core fuel bundles are still intact and the Zircaloy clad oxidation is about to start. The objective of the study is to provide the conditions of reflood into the hot core due to the level swell or a slug delivered from the lower head as the control rod melt drops into the water. These conditions will be employed in the neutronic analysis with the RECRIT code to determine if the core recriticality may be achieved.

The interaction of the control rod melt with the water in the lower head assumed:

- a non-energetic melt-coolant interaction (MCI), and
- an energetic MCI

The analysis performed was based on very conservative assumptions and employed hand calculations and engineering judgement, based on many years of experimental and analysis-development research. The following results were obtained:

Non-Energetic MCI Reflooding Parameters

Steam flow rate through the reactor core = 8m/s.

Void fraction of in-core coolant = 30%

Overall level swell (core reflood) =0.33 m

Energetic MCI Reflooding (Slug Penetration) Parameters

- Slug entering the whole cross section area of core
    - RAPID PENETRATION
      Velocity of slug = 60 m/sec
      Period of slug penetration = 7 ms
      Void fraction of slug = 0%
      Total coolant volume in core = 1.5 m^3
      Reflood height ~0.4 m
    - SLOW PENETRATION
      Velocity of slug = 20 m/sec
      Period of slug penetration = 20 ms
      Other parameters the same as above
- Slug entering the peripheral half of the core cross sectional area
    - RAPID PENETRATION
      Velocity of slug penetration = 80 m/s
      Period of slug penetration = 10 ms

Void fraction of slug = 0%
Total coolant volume in core = 1.5 m^3
Reflood height = 0.8 m

- SLOW PENETRATION
  Velocity of slug penetration = 20 m/s
  Period of slug penetration = 40 ms
  Other parameters the same as above.

A further study was made for the case when the severe accident had progressed further and corium melt would be available for entry into the water in the lower head. A steam explosion or a level swell for that point of time in the scenario was considered. However, it was argued that a large explosion is unlikely in the BWR lower head due to the presence of a forest of control rods. Additionally, a level swell or a slug of water would probably not have any easy access to the core because of the blockages formed. It was also argued that with the loss of corium from the core, the core would be highly subcritical. All of these conditions argue against the likelihood of a core recriticality. It is recommended, however, that if further confidence in the avoidance of criticality of the core (or of a reactivity initiated accident) is desired, mechanistic analysis of the steam explosion process in the BWR lower head, coupled with the formation of blockages at the core-plate and the subcriticality of the core due to corium removal should be performed. Such an analysis, however, will not be easily developed and performed.

The Possibility and the Effect of a Steam Explosion in the BWR Lower Head on Recriticality of a BWR Core

1. Background

The subject matter of this report is the assessment of the possibility of a reactivity induced accident (RIA) accompanying a conventional loss of heat removal scenario during a severe accident in a BWR. The scenario for such an event is postulated as follows:

A series of initiating faults lead to the conditions in a BWR during which the automatic depressurization system (ADS) is activated but no water sources are available. Such a scenario, possible for a station black-out event, could lead to the so called 'dry core' condition in which the core water level has dropped below the core plate. The reactor has been scrammed, however, the residual and the decay heat cause the core to heat up sufficiently to cause Zircaloy clad oxidation.

The first disruption of the core geometry occurs when the core temperatures reach ~1000°C and the cruciform control rods situated between each set of four rod bundles start liquefying, due to the formation of a eutectic between the B_4C and the stainless steel [Hoffman]. The control rod melt accumulates on the core plate. The construction of the core plate on top of the control guide tube (CRGT) is such that the control rod melt would most likely drop into the housing of the CRGT, where it will not have enough water to cause a steam explosion. However, there are other possibilities, i.e. the drop of the control rod melt into the water in the lower head. This could occur in three different modes. (a) the control rod melt goes through the small openings around the rod bundles, into the lower head, or (b) the control rod melt eats through the Zircaloy shroud and the water entry orifice and drops into the lower head, or (c) the control rod melt accumulates on the core plate, heats up the plate to high enough temperatures so that the core plate suffers a creep failure and the accumulated control rod melt drops en-mass into the water contained in the lower head. During this time interval the core heat-up may have progressed to the melting of the Zircaloy clad however the UO_2 pellets are intact in the core. The core bundles, at that point in time, are basically intact, except that they are at high temperature.

The drop of the molten eutectic mixture of B_4C and stainless steel into the water in the lower head may lead to a steam explosion. It may be postulated that either, (i) the steam explosion does not occur but the large and rapid steam generation leads to a level swell which enters the rod bundles at a certain velocity and to a certain level in the core or (ii) the steam explosion occurs, generating energy to drive a slug of water into the core bundles. Both of these postulated events may lead to an RIA, since the core is basically devoid of the B_4C control rods, which were initially in the core when the scram was activated. The most reactive state for the core would be at the beginning-of-life for a core cycle. Clearly, in this scenario, although the core is in its most reactive state, the potential for steam explosion with the B_4C-steel melt is rather low, since (1) the melt mass is not large and (2) the melt temperatures may be below the steel melting temperature of 1400°C, i.e., there may be very low superheat.

Another scenario may be of concern, which occurs later in the accident progression. In this scenario the core is devoid of the control rod material which may be on the core plate or in the lower head, after it dropped into water in an incoherent mode without creating substantial level swell or a steam explosion. The accident proceeds and in time a substantial accumulation of the core melt occurs (melt pool) in the BWR core as it did in the TMI-2 accident. The in-core melt pool may then drop into the lower head from the side of the core as it did in the TMI-2 accident or it could break through the blockage at bottom. This could also create either a level swell or a steam explosion generated water slug entry into the rod bundles. An RIA is possible also in this case, however, the reactivity addition has to overcome the reactivity loss due to the removal of UO_2 and Zircaloy from the core. Clearly in this case the core is less reactive, however the potential for a steam explosion is higher due to the entry of a larger quantity of melt, at high superheat, into the water contained in the lower head. This scenario should be investigated to discover if there is a time-window when the core has not lost substantial amount of fuel and clad and a steam explosion induced water slug entry into the core can be postulated. Such a scenario is also possible for a PWR severe accident in which the silver-indium-cadmium control rods also melt early and their melt also accumulates in the lower regions of the core bundles, lodged between the core blockage and the water level.

Steam explosion research has never considered modelling the lower head geometry of a BWR with its forest of CRGTs and instrumentation tubes. As mentioned above there are not large size bodies of water in between the CRGTs and any large diameter corium melt drop may not have enough water available to make a pre-mixture capable of propagating as a steam explosion. Currently, with a lack of database and analysis results, we can only take the path of highly conservative assumptions which may be highly unrealistic.

2. Previous Studies

The suspicions, that entry of a two phase mixture or a slug of water into the BWR core devoid of control rods may create a reactivity spike, led to three previous studies on this issue in Finland.

Antilla (VTT, 1990) performed neutonic analysis for the TVO core without control rods and with water or two phase steam-water mixture addition. He found that for such a core, the solid water (void fraction 0%) level has to reach at least 0.4 m to pose a threat of promp-criticality. For a steam-water mixture with void fraction of 60%, the two phase mixture level has to reach 1 meter for the threat of promp-criticality.

Okkonen, Hyvarinen and Haule (1993) evaluated the recriticality potential for a 2200 MWt BWR due to an FCI in the lower head. Their analysis considered only non-energetic FCIs and the approach was that if there is a potential of prompt recriticality with non-energetic FCIs. Then there may be a greater potential with energetic FCIs. They chose a certain mass of melt, the heat transfer correlations and employed RELAP-5-Mod 2 to calculate the core reflood due to the penetration of two phase mixture into the core. The chosen melt mass was very large, i.e., 4-8 tonnes in a very well fragmented state (1mm particle size diameter). The energy delivered was 200 to 25 000 MW during 0.5 to 12 seconds. The steam produced generated a slug of water entering into the core which was sufficient to cause prompt criticality. The analysis was performed in one dimensional geometry and no blockages were considered at the core bundle inlet orifices. The core melt, degradation and melting delivery scenarios were ignored. We shall provide a critique of the Okkonen, Hyvarinen and Haule study later in this report.

Recently study of the potential for recriticality for a BWR core devoid of control rods has been performed by Frid at al. in the SARA Project performed in the Fifth Framework program of the European Union. This study was not concerned about the FCI process in the BWR lower head but instead with the process of core reflood during the early part of a postulated severe accident when the control rods have melted but the fuel rod bundles are intact. The ECCS water injected is cold and unborated and its entry from the core bottom was modelled to determine the potential for prompt criticality. Core reflood analysis was performed and it was calculated that for large rates of reflood $\geq$ 500 Kg/sec, the potential for a super prompt criticality exists. The core criticality was terminated with the Doppler feed back, however, substantial energy was deposited in the fuel rods. Continuation of the reflood may lead to the core reaching an almost steady state power level of between 10 and 20% of the nominal power, which, if sustained, could lead to the boiling of the suppression (condensation) pool and containment pressurization.

Recriticality studies of the BWR without control rods have been reported by Shamoun and witt (1994), Mosteller and Rahn (1991), Bandurski et.al. These studies showed that for (i) realistic reflood rates, (ii) void fractions of > 20% and (iii) retention of $\geq$ 20% of control rod material in the core, prompt recriticality would not occur. Core power levels of ~10% may be achieved if reflood continues to fill the core. These analyses employed single rod and one-dimensional geometry and three-dimensional effects were not treated.

3. Objectives of the Present Study

The main objectives of our study is to provide the initial conditions of reflood, into a hot (~1800 K), dry BWR core without control rods, from the lower head pushed in by:

- a non energetic FCI and
- an energetic FCI

in order to evaluate the potential for prompt recriticality. The neutronic criticality calculations will be performed with the code RECRIT by VTT. The initial conditions of interest are the rate and void fraction of the reflood at core entrance.

4. Evaluation Approach Employed

We consider first the BWR dry core scenario and consider the time window when the temperature of the core is above ~1200°C, i.e. when the B_4C-steel cruciform, control rods have melted, while the fuel rods are intact. We will consider the addition of control rod melt into the saturated water in the lower head to determine (a) the potential for a steam explosion and (b) the characteristics of the reflood,

Second, we will consider the above scenario but assume that an energetic FCI occurs for certain fuel drop conditions,
Third, we will further examine the parameters for the occurrence of an energetic FCI,

Fourth, we will reexamine the assumptions employed by Okkonen et al. in their study for the potential of prompt recriticality and finally we will provide the initial conditions for the water-steam mixture reflood to the core for the non-energetic and energetic FCI cases. Most of the results provided are based on our engineering judgement and hand calculations. The BWR dry core scenario conditions calculated with the MELCOR code will be employed in our evaluations.

5. Initial Conditions

5.1 MELCOR Calculations for BWR Dry Core Scenario

A set of MELCOR 1.8.4 Code calculations were performed by Ilona Lindholm of VTT Energy in connection with the E.U.'s SARA Project and provided to us. The BWR scenario considered is that of Station black-out with successful depressurization leading to a dry core. The options employed in the core allowed the melting of the cruciform control rods. The core was divided into 5 radial rings with equal cross section area and 25 axial levels. The total masses in the each of the inner four rings of core region/radial rings were as follows:

B_4C	~ 252 Kg
Steel	~ 3000 Kg
UO_2	~ 20500 Kg
Zr	~ 6700 Kg

Risto Sairanen of VTT Energy recommended to us to employ the MELCOR output at the time of 4700 seconds when the B_4C-steel control rods had melted and had settled on the core support plate. At that point in time in the core heat up process, except for the outer fifth radial ring, all the steel and B_4C had been collected on the core support plate. The total accumulations available for relocation were B_4C = 1131 Kg and steel = 13189 Kg. (See Table 1, in which the support plate is node 4 and the active core extends from node 5 to 29). The temperatures of control rod materials at 4700 seconds in the radial rings varied from 1681K in the center ring to 1497K in the outer radial ring (see Table 2). It should be noted that these temperatures are lower than the melting point of stainless steel. The calculated temperatures for the support plate at 4700 seconds vary from 1203K to 660K. The stainless steel material of the support plate can creep at 1203K, however, creep deformations to failure could occur only with applied pressures of 25 bars or higher. At 4700 seconds the weight of the control rod material deposited on the support plate is not sufficient to induce creep failure. Support plate temperatures have to rise much higher before the potential for creep failure could become significant. MELCOR calculated results at t = 4700 seconds for the component and coolant temperatures in the five rings are shown in Table 3. These are the average temperatures for the components, i.e. (UO_2+ ZrO_2+ Zr) in the core bundles contained in the five rings. The MELCOR predicted temperatures are still substantially below the Zircaloy melting temperature. Thus, at that point in time, the fuel rods are still intact, although at quite high temperature. The steam coolant is at the same temperature as the fuel rods although it should be at a somewhat lower temperature.

5.2 VESSEL, CRGT and Fuel Bundle Geometries

The geometries of the vessel, CRGT, control blade, fuel bundle and the associated locations and flow areas are important in determining the control rod melt and later the corium melt discharges to the water in the lower head. Fig. 1 shows the overall schematics of the vessel with the arrangement of control rods and the fuel assembly. Fig. 2 shows the schematic detail of the core plate, CRGT and the moderator (coolant) flow paths into the 4 fuel rod bundles which plug into the construction at the top of each CRGT. Fig. 3 provides further detail of the flow paths leading to each fuel bundle and the bypass flow in-between the neighbouring bundles. These 3 figures show the configurations for a General electric designed BWR. Figure 4 shows the vessel, CRGT and the details of the lower head for the vessel in the TVO BWR designed by ASEA Atom (later ABB). The top view of the placement of the CRGTs and the

Table 1 Steel and B4C masses (kg) available for relocation at t=4700 s. Node 1 = bottom, node 30 = top.

Level	Ring 1, SS	Ring 1, B4C	Ring 2, SS	Ring 2,B4C	Ring 3, SS	Ring 3,B4C	Ring 4, SS	Ring 4,B4C	Ring 5, SS	Ring 5,B4C
30	0	0	0	0	0	0	0	0	0	0
29	0	0	0	0	0	0	0	0	0	0
28	0	0	0	0	0	0	0	0	0	0
27	0	0	0	0	0	0	0	0	0	0
26	0	0	0	0	0	0	0	0	0	0
25	0	0	0	0	0	0	0	0	0	0
24	0	0	0	0	0	0	0	0	0	0
23	0	0	0	0	0	0	0	0	0	0
22	0	0	0	0	0	0	0	0	0	0
21	0	0	0	0	0	0	0	0	0	0
20	0	0	0	0	0	0	0	0	0	0
19	0	0	0	0	0	0	0	0	0	0
18	0	0	0	0	0	0	0.92	0	0	0
17	0	0	0	0	0	0	2.59	0	0	0
16	0	0	0	0	0	0	4.55	0	0	0
15	0	0	0	0	0	0	7.71	0	0	0
14	0	0	1.23	0	4.618	0	13.58	0	0	0
13	11.58	0	12.88	0	16.69	0	25.28	0	0	0
12	0.52	0	0.54	0	0	0	30.22	0.019	0	0
11	11.85	0.014	15.74	0.019	27.12	0.037	53.98	0.081	0	0
10	40.59	0.086	45.56	0.102	56.41	0.134	69.72	0.158	7.96	3.53
9	53.48	0.154	57.31	0.172	67.22	0.214	77.46	0.26	15.3	6.801
8	59.9	0.222	64.05	0.251	73.11	0.314	80.69	0.444	20.91	9.295
7	25.3	0.065	43.64	0.172	68.23	0.388	86.32	1.159	28	12.45
6	59.85	0.409	66.97	0.523	80.42	0.913	76.82	2.119	26.31	11.72
5	428.75	12.74	378.9	13.59	315.9	17.97	256.2	32.77	108.1	48.17
4	2604.6	237.96	2605.5	236.76	2575.5	231.66	2248.1	214.56	74.91	33.3
3	0	0	0	0	0	0	0	0	0	0
2	0	0	0	0	0	0	0	0	0	0
1	0	0	0	0	0	0	0	0	0	0
	3296.43	251.65	3292.3	251.589	3285.238	251.63	3034.17	251.57	281.49	125.266
Steel (kg)	13189.63									
B4C (kg)	1131.705									

Table 1

Table 2 Control material temperatures (K) in core nodes at t=4700 s. Node 1 = bottom, node 30 = top

Table 2

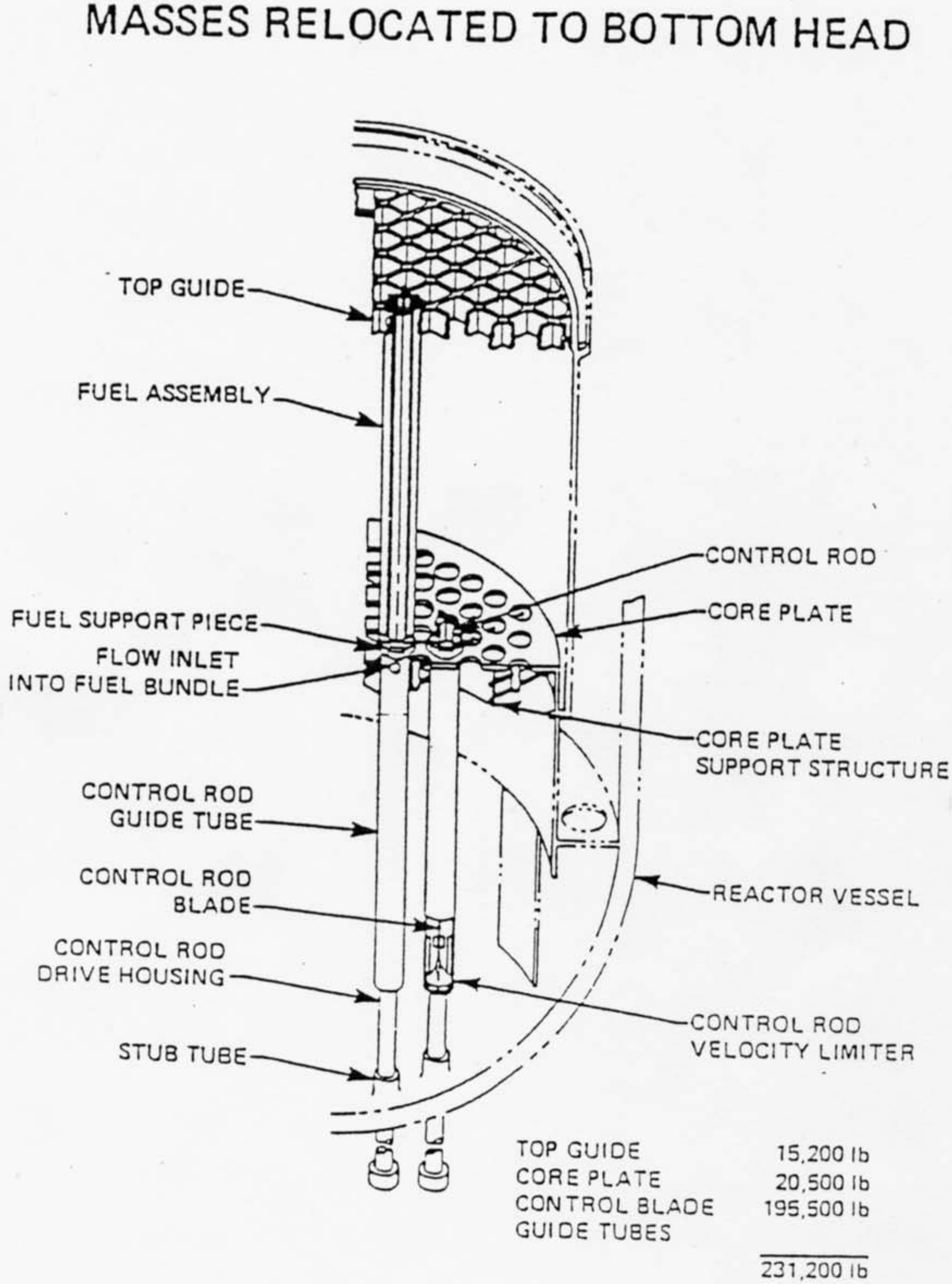
Level	Ring 1	Ring 2	Ring 3	Ring 4	Ring 5
30					548
29					536
28					578
27					701
26					1022
25					1519
24					1633
23					1666
22					1670
21					1663
20					1653
19					1642
18				1700	1631
17				1700	1621
16				1700	1610
15				1700	1595
14		1700	1700	1700	1569
13	1700	1700	1700	1700	1519
12	1700	1700	1700	1699	1519
11	1698	1698	1698	1698	1514
10	1698	1698	1697	1697	1498
9	1697	1697	1697	1696	1498
8	1696	1696	1696	1695	1498
7	1697	1696	1694	1687	1498
6	1695	1694	1691	1676	1502
5	1681	1673	1658	1613	1497
4	***1203***	***1204***	***1185***	***1104***	***660***
3	394	393	391	389	386
2	384	384	384	384	384
1					

Table 3 Component and coolant temperatures (K) in core nodes at t=4700 s. Node 1 = bottom, node 30 = top.

Level	Ring 1, Comp	Ring 1, Cool	Ring 2, Comp	Ring 2, Cool	Ring 3, Comp	Ring 3, Cool	Ring 4, Comp	Ring 4, Cool	Ring 5, Comp	Ring 5, Cool
30	522	522	522	522	521	521	524	526	532	504
29	532	532	532	532	532	531	531	531	531	530
28	577	576	577	576	577	575	575	574	568	566
27	734	733	734	733	731	730	723	722	686	682
26	1176	1178	1174	1176	1167	1169	1140	1142	1030	1032
25	1761	1760	1757	1756	1743	1742	1694	1694	1555	1560
24	1906	1905	1901	1900	1882	1882	1821	1821	1659	1661
23	1955	1954	1949	1948	1929	1928	1861	1860	1694	1695
22	1965	1964	1958	1957	1938	1937	1867	1867	1698	1700
21	1956	1955	1950	1949	1929	1928	1856	1856	1693	1695
20	1941	1940	1934	1933	1913	1913	1841	1841	1683	1685
19	1923	1922	1917	1916	1896	1896	1824	1824	1673	1675
18	1905	1905	1900	1899	1880	1880	1811	1815	1663	1666
17	1890	1889	1884	1883	1865	1865	1805	1810	1655	1657
16	1874	1873	1869	1868	1850	1850	1799	1805	1646	1649
15	1856	1856	1851	1850	1833	1833	1792	1799	1636	1639
14	1831	1833	1828	1831	1818	1823	1784	1792	1622	1627
13	1818	1825	1816	1823	1806	1815	1773	1782	1605	1615
12	1806	1817	1803	1814	1794	1805	1761	1771	1595	1605
11	1794	1807	1791	1804	1782	1795	1749	1758	1581	1592
10	1782	1796	1779	1793	1770	1783	1736	1744	1569	1583
9	1772	1786	1769	1783	1759	1771	1724	1730	1561	1574
8	1764	1777	1760	1772	1749	1759	1712	1716	1553	1565
7	1766	1776	1756	1766	1740	1748	1698	1700	1542	1552
6	1760	1771	1751	1761	1730	1736	1685	1685	1532	1536
5	1713	1731	1700	1713	1646	1635	1590	1573	1457	1430
4	1061	879	1062	880	1029	837	955	785	660	569
3	394	409	393	408	391	406	389	401	386	398
2	384	384	384	384	384	384	384	384	384	384
1		384		384		384		384		384

Table 3

Fig. 1


MASSES RELOCATED TO BOTTOM HEAD
TOP GUIDE
FUEL ASSEMBLY
CONTROL ROD
CORE PLATE
FUEL SUPPORT PIECE
FLOW INLET INTO FUEL BUNDLE
CORE PLATE SUPPORT STRUCTURE
CONTROL ROD GUIDE TUBE
CONTROL ROD BLADE
REACTOR VESSEL
CONTROL ROD DRIVE HOUSING
CONTROL ROD VELOCITY LIMITER
STUB TUBE
TOP GUIDE 15,200 lb
CORE PLATE 20,500 lb
CONTROL BLADE GUIDE TUBES 195,500 lb
231,200 lb

FUEL ASSEMBLY SUPPORT COMPONENTS

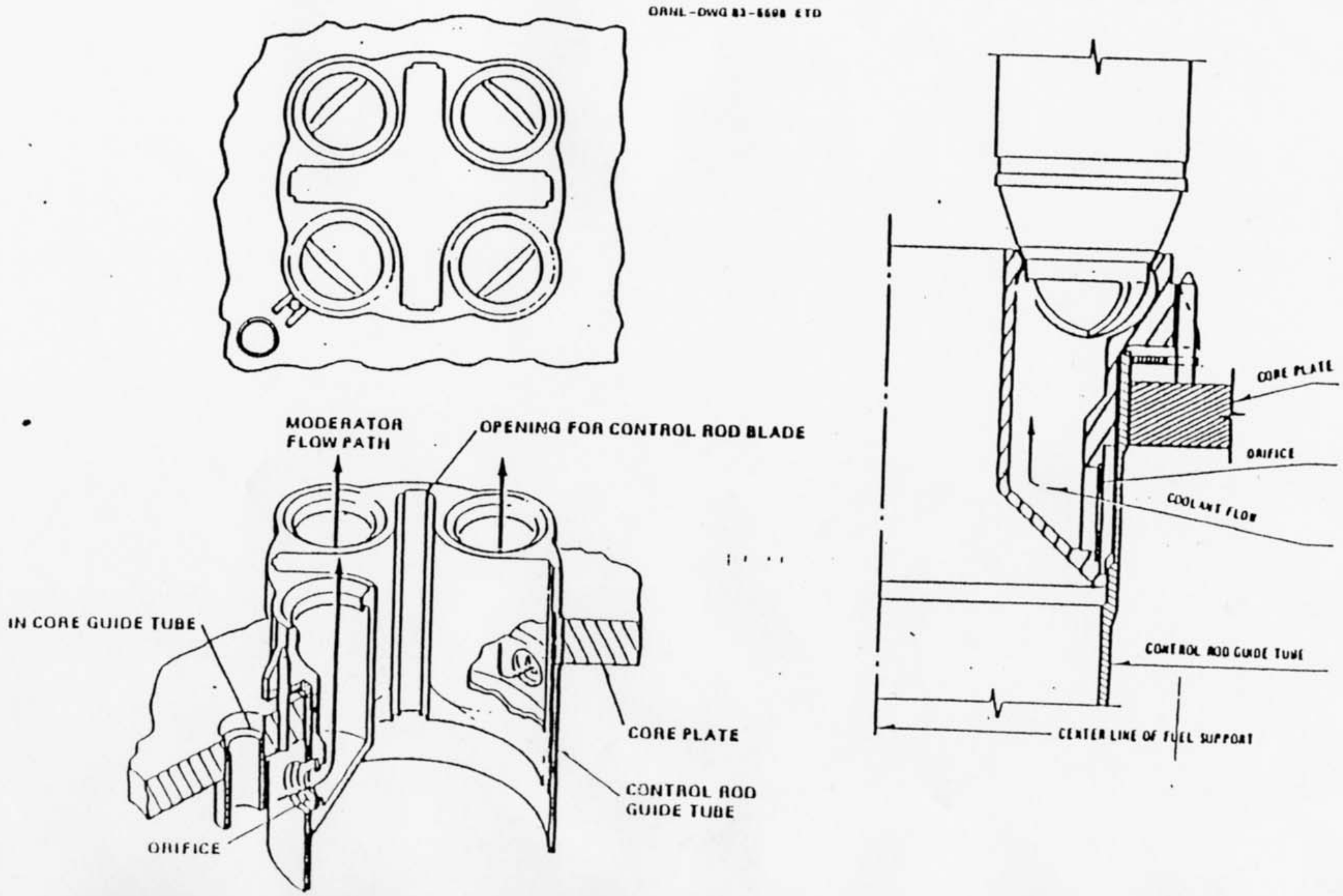
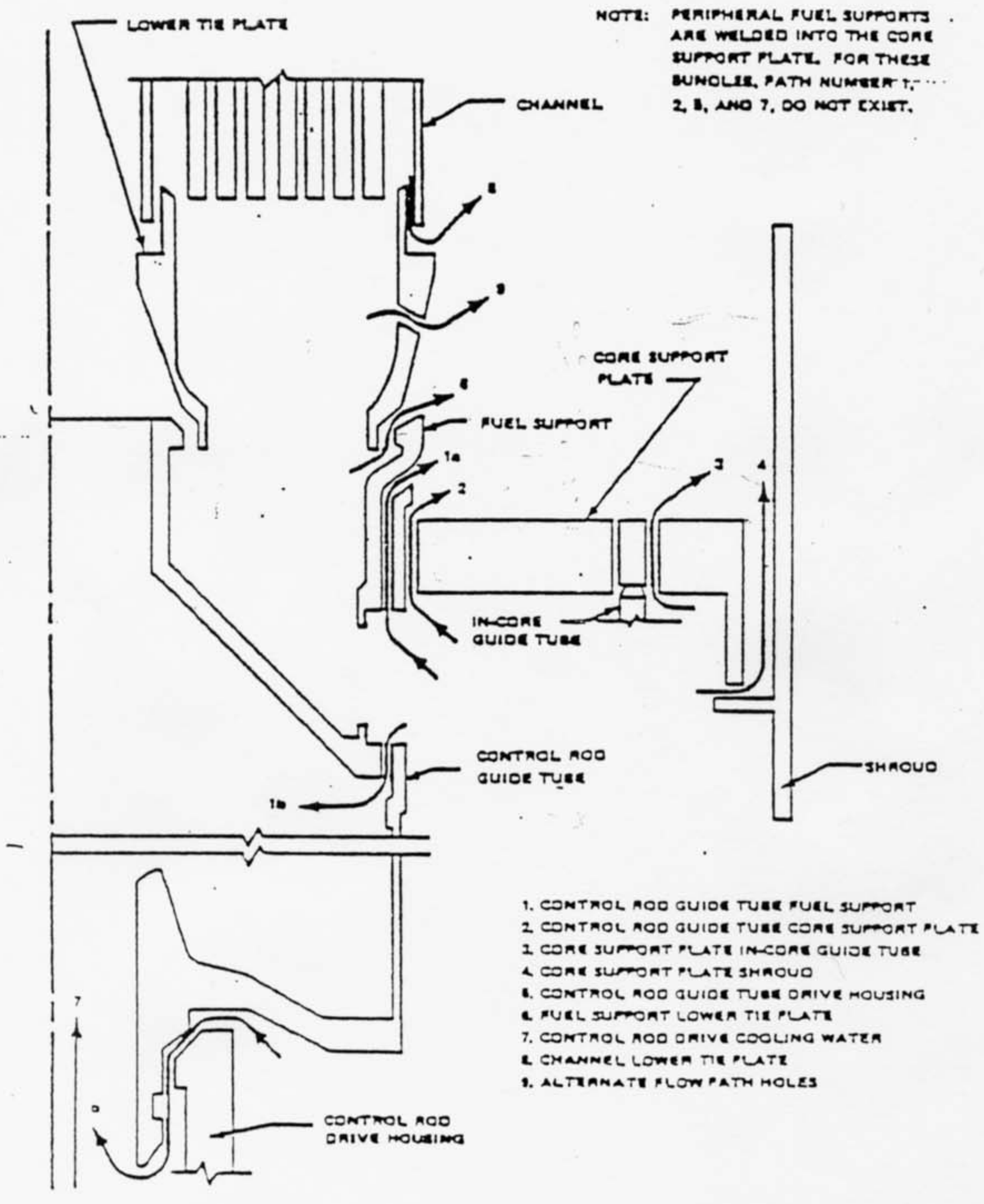


Figure 2

Figure 3


NOTE: PERIPHERAL FUEL SUPPORTS ARE WELDED INTO THE CORE SUPPORT PLATE. FOR THESE BUNDLES, PATH NUMBER 1, 2, 5, AND 7, DO NOT EXIST.
LOWER TIE PLATE
CHANNEL
CORE SUPPORT PLATE
FUEL SUPPORT
IN-CORE GUIDE TUBE
CONTROL ROD GUIDE TUBE
SHROUD
CONTROL ROD DRIVE HOUSING
1. CONTROL ROD GUIDE TUBE FUEL SUPPORT
2. CONTROL ROD GUIDE TUBE CORE SUPPORT PLATE
3. CORE SUPPORT PLATE IN-CORE GUIDE TUBE
4. CORE SUPPORT PLATE SHROUD
5. CONTROL ROD GUIDE TUBE DRIVE HOUSING
6. FUEL SUPPORT LOWER TIE PLATE
7. CONTROL ROD DRIVE COOLING WATER
8. CHANNEL LOWER TIE PLATE
9. ALTERNATE FLOW PATH HOLES

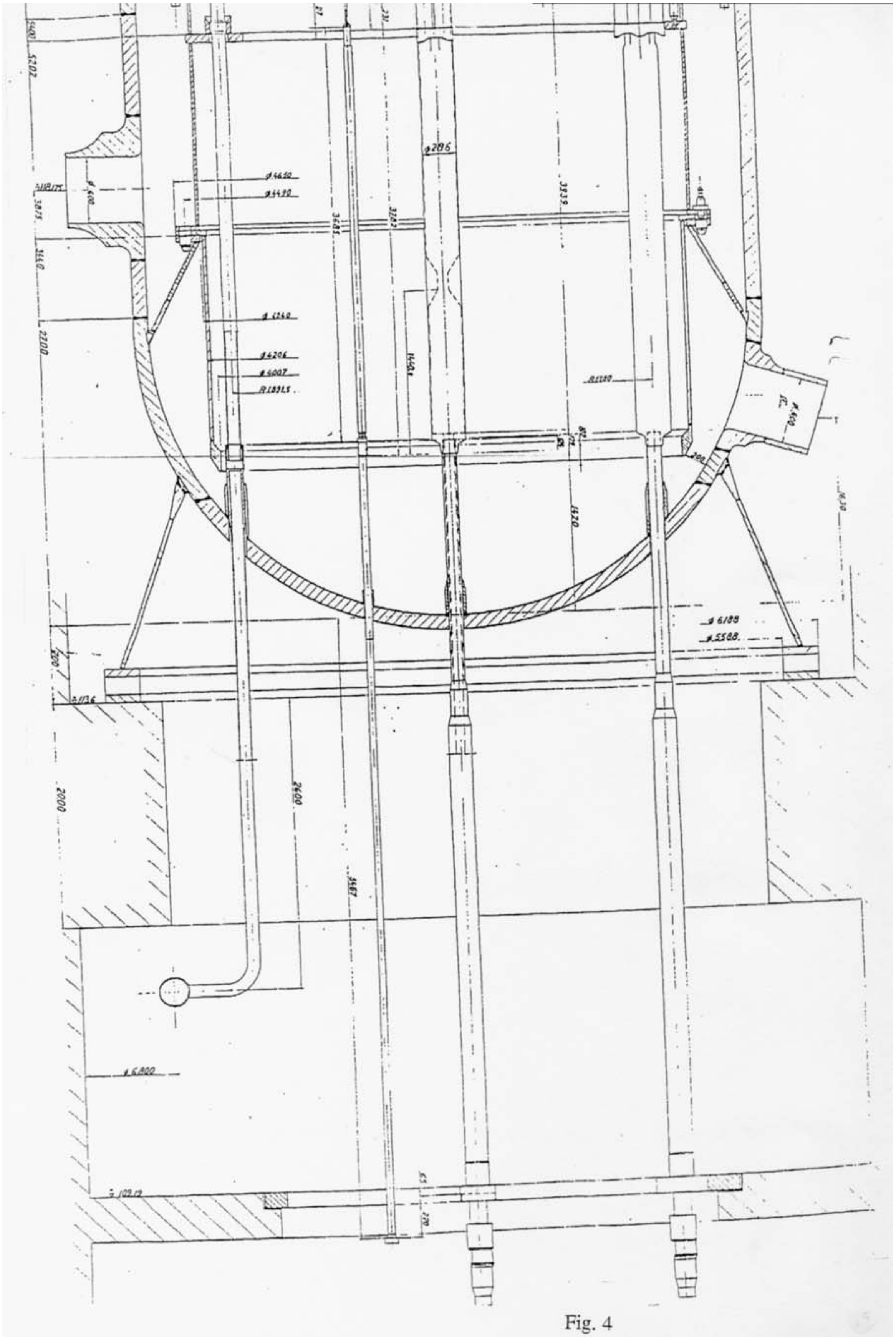


Fig. 4

neutron flow detector nozzles is shown in Fig. 5; and Fig. 6 provides the details on the CRGT diameter and pitch in the vessel in meters. Figure 7 provides the flow areas around a control rod cruciform blade and for the coolant in a fuel bundle. From the drawings for the TVO vessel, we derive that there are 121 CRGTs, each located at the pitch of 30.75 cm with the CRGT outer diameter of 12.45 cm. In-between the four CRGTs, there is an instrument tube, located in the body of water, which has an area of 0.08 m^2 and depth of ~3 meters, i.e. a volume of ~0.24 m^3. The flow areas around each control rod is 0.006 m^2, which provides the direct path for the control rod melt to enter the CRGT annulus in which it could freeze and block the channel. If freezing does not occur, the control rod melt could discharge into the lower head water through holes in the CRGT.

The other pathway for the control rod melt flow is through the coolant orifice in the rod bundle whereby it can reach the water in the lower head. The flow area for this pathway is 0.008 m^2. However, the melt has to eat through the Zircaloy shroud around the fuel rods.

Clearly, the BWR lower head geometry, populated with a forest of CRGTs is not as open as the geometry of the PWR lower head. Thus, it is not so conducive to the formation of an efficient pre-mixture, a precursory condition for propogation to a steam explosion. The CRGTs should also serve as heat sinks for any fragmented melt particles that touch them.

6. Analysis for Core Reflood Characteristices due to Melt-Water Interactions (MCI)

As indicated in the evaluation approach, this analysis will be based on engineering judgement and hand calculations using the initial conditions of melt accumulations and the geometry of the vessel. We will divide the analysis into the following tasks:

Task 1: Control Rod Melt drop into lower head water without occurrence of a steam explosion
Task 2: Same as above, except a steam explosion is postulated to occur
Task 3: Evaluation of other scenarios, which may lead to steam explosion and recriticality.

The analysis and evaluations performed in each of these tasks are reported in the following paragraphs. It should be mentioned that we will be employing highly conservative assumptions. The Conclusions and Recommendations are provided in the final section of the report.

6.1. Task 1 <u>Control Rod Melt Drop into Lower Head Water without Occurrence of a Steam Explosion</u>

<u>6.1.1 Evaluation Melt Configuration Prior to Release and the Melt Discharge Conditions</u>

We will first limit our considerations to a non-energetic interaction of the control rod melt with the water in the lower head of the BWR. For initial conditions we employ the accumulated control rod melt characteristics calculated by the MELCOR code shown in Tables 1 and 2. Approximately 1130 Kg of B_4C and ~13000 Kg of steel melt are available for release to the lower head. Taking account of the densities of B_4C of 2500 Kg/m^3, of steel-B_4C eutectic of 3800 Kg/m^3, of steel of 7500 Kg/m^3, approximately 2.1m^3 of melt is available for release. The total area of the core plate is ~12m^2, however the cruciform area is ~18%. Not all

Figure 5

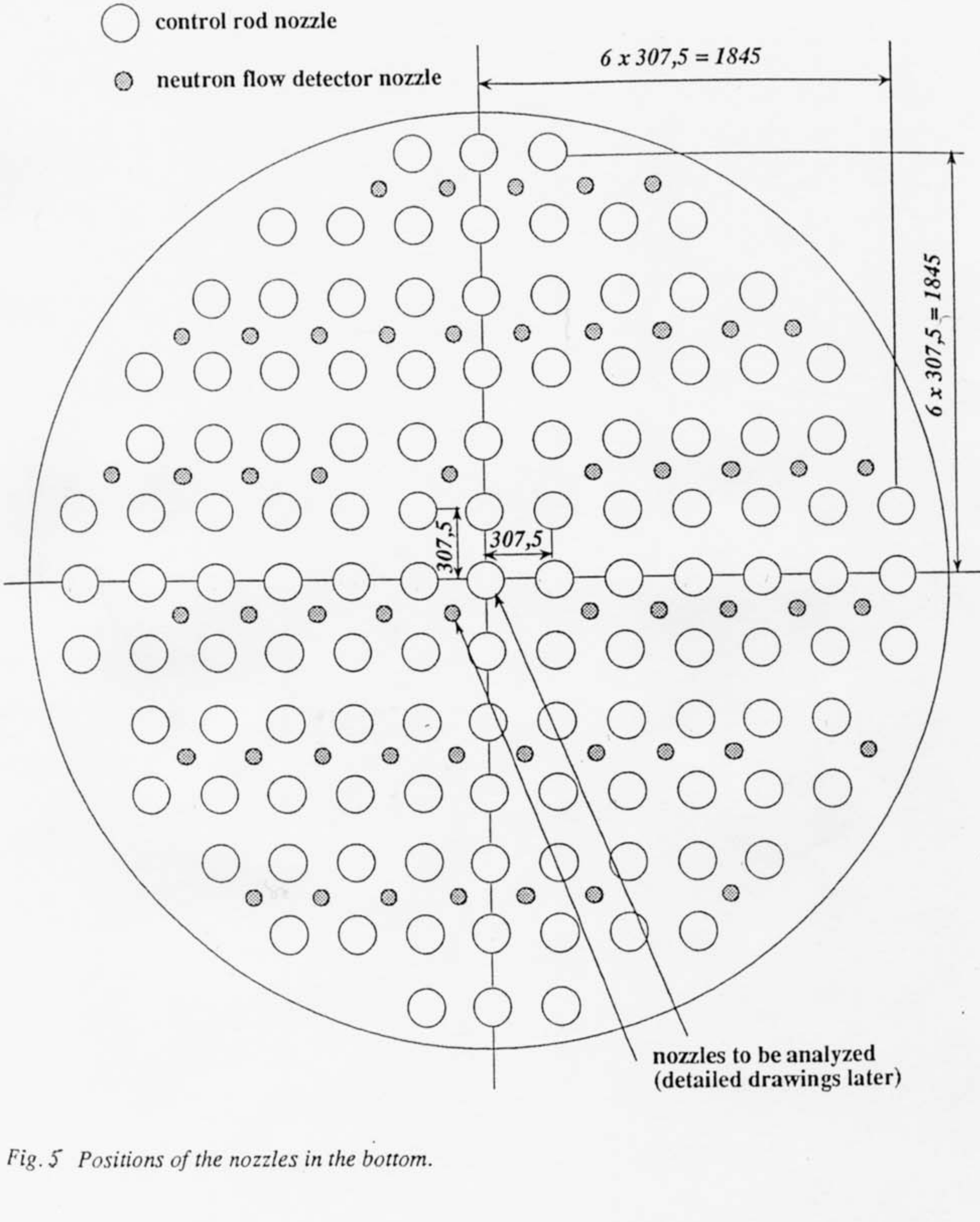


Fig. 5 Positions of the nozzles in the bottom.

Figure 6

	A	B	C	D	E	F	G	H	I
1									
2									
3									
4									
5									
6				Above the crd nozzle		Below the CRD nozzle			
7									
8		Pitch		0.3075		0.3075			
9		Crd tube outer diam		0.1245		0.161			
10		Area per tub		0.082382		0.074198			
11									
12		Amount of water within the tube							
13		Crd tube inner diameter		0.1					
14		Piston tube outer diameter		0.069					
15		Area water		0.004115					
16									
17									
18									
19									
20									
21									

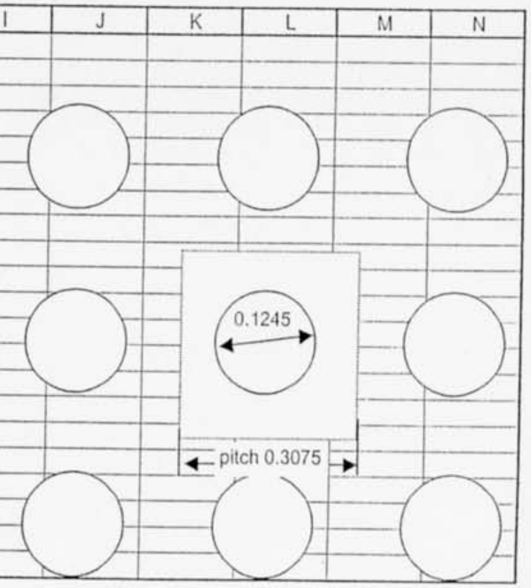
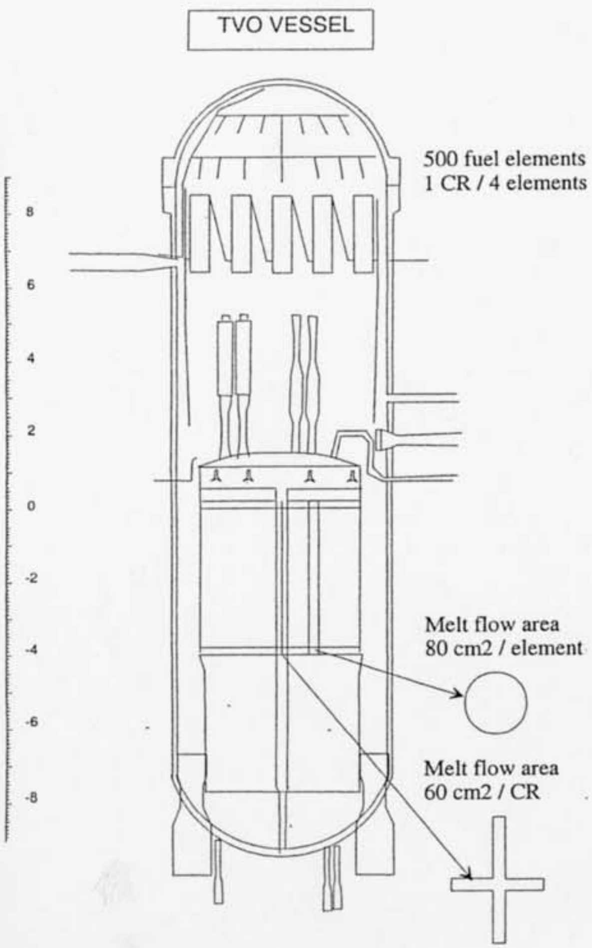


Fig. 6

Real vessel geometry and flow area relations



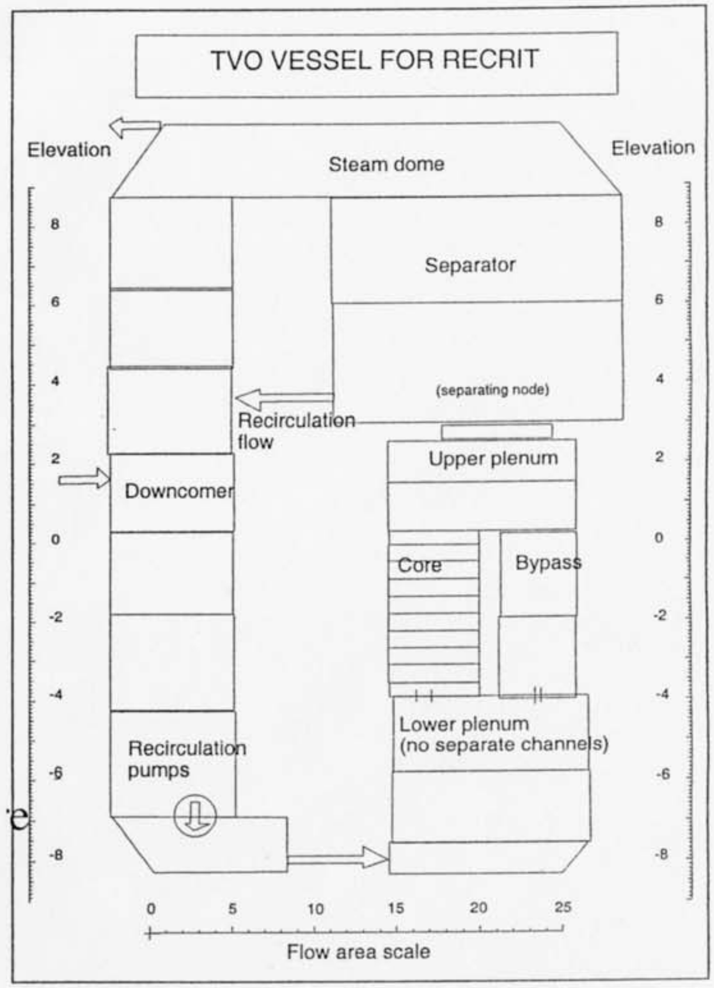


Figure 7

VTT ENERGY

the core plate will be uniformly loaded by the control rod melt. Thus, the height of the melt in the cruciform areas could be of the order of ~1m. Assuming that the water level is ~0.1 meters below the bottom of the core plate, an effective hydraulic head of ~1.6m could prevail for the flow of the control rod melt. This would provide a melt drop velocity of 5-6 m/sec, which will gradually diminish as the melt pool level on the core plate drops.

The melt is ejected through a hole whose maximum size is determined by the cross section area of the cruciform and it is about 60 cm^2. No hole ablation will take place since the melt temperature is lower than the melting point temperature of the structural steel. The melt discharge hole will most likely be located underneath the core central region where the melt temperature is the highest. This would be the most likely location for the melt flow, since the Zircaloy shroud on the rod bundle is still intact at t=4700 seconds.

Considering the above analysis, we obtain that:

The maximum flow rate for the melt into the CRGT will be 0.033 m^3/sec. or about 200 Kg/sec.

This implies that the 14000 Kg of accumulated melt can be deposited to the lower head in $\geq$ 70 seconds. During this period the melt-water interaction process in the lower head will produce steam, whose upward flow will reduce the melt drop rate.

This is a substantial relocation flow rate, of the similar order as considered by Theofanous and co-authors for the evaluation of the steam explosion loading of the lower head of the AP-600 vessel. In this case, however, the low melt temperatures for its composition of steel and B_4C are not very conducive for a steam explosion.

The 14000 Kg of melt must have a crust at the bottom and top of the melt pool, for it to exist as melt, or for it to drop into the melt pool as a coherent mess. In the absence of a crust, the B_4C and steel alloy would be dropping gradually as it arrives on the core plate. Thus, the assumption of a coherent drop of the control rod melt is highly conservative.

6.1.2 Evaluation of Melt-Coolant Premixing and Core Reflooding

The melt (steel and B_4C) jet of ~9 cm diameter enters the water pool at 5-6 m/s. The break-up length can be estimated by Saito correlation, which results in 1.2 m. Since the melt relocation, most probably, is in the central region, a significant fraction of the jet will fragment into droplets. The situation is very similar to the FARO experiments for saturated water pool, which show the typical droplet size d_p = 3-5 mm. The premixing zone (melt-coolant interactions) is about 1 m in diameter and goes down from the water surface to the pool bottom. The mixing volume of approximately 3 m^3 can be envisioned.

The heat transfer from the melt to coolant can be evaluated from the radiative and the film-boiling components. For emissivity of 0.45, the radiative heat flux is calculated to be 200 kW/m^2 for 1700 K, and the film boiling is 270 kW/m^2. In sum, the heat removal is much less than the CHF value of $1MW/m^2$. Note that in the pool, the melt temperature cools down so the heat flux goes down to 300 kW/m^2, i.e. essentially film boiling, since the temperature remains greater than the minimum film-boiling temperature.

The amount of melt in the mixing zone is evaluated to be M = 1200 kg, i.e. it takes effectively 6 seconds for the melt to sediment in the pool. The heat removal rate is evaluated to be

$$Q = 6\, q''\, M/(d_p\, \rho_m) \;\sim 120 \text{ MW}$$

The steam production rate is Q/h_{fg} = 55 kg/s or 50 m^3/s at 2 bar system pressure. This steam will vent through the core and the pump (making the pump to work like turbine). Through the core, it will go into the fuel channels and unblocked inter-channel spaces. Given the cross-section area of the lower head is 15 m^2, the steam velocity is 3.2 m/s. For the core region, the steam velocity will be about 8 m/s.

Given the rate of energy supply to the pool by steel –B_4C melt is 200 kg/s x 0.8 MJ/kg ~160 MW, only 75% of this is extracted during the premixing phase. The rest is stored in the melt/debris accumulated on the lower head bottom. For steel latent heat (0.32 MJ/kg) and specific heat (0.4 kJ/kg.K), this 75% heat extraction indicates that the melt is solidified, and cools down from 1700 K to about 1000 K.

From the FARO experiments, as well as from results of premixing analyses, it can be seen that the level swelling is about 0.5 to 1 m for small pool (FARO), and far less in the large-diameter BWR lower head. In fact, given the premixing zone volume of 3-4 m^3 with its effective void fraction of 50% the level swell in the core area (cross-sectional area of 12 m^2) will be ~0.3 m.

If the coolant level just prior to MCI is ~0.1 m below the core plate, then, the non-energetic MCIs will cause an in-core level-swelling. As the coolant enters the core, it will further vaporize due to intense heat removal upon rewetting (~1 MW/m^2). Given the in-core void fraction typical for nucleate boiling (30%), the in-core level swelling could be ~0.33 m.

As the water from the lower head goes through the core plate, it would cool the remaining control rod melt on the core plate and change the characteristics of the subsequent melt drop. It should be noted that the melt drop time of 70 seconds is a considerably long time and it would become much longer with the scenario as described and calculated above, due to the intense steam flow upwards from the water surface.

The above scenario also assumes that there are no blockages of the holes in the core plate for bypass flow or of the orifices in the fuel bundles.

Thus, for the non-energetic control rod melt-water interaction in the lower head of the BWR, the following conditions of the core reflood apply:

Steam flow through the reactor core	***= 8m/sec***
Void fraction of the in-core reflood coolant	***= 30%***
Overall core reflood swell level	***=0.33 m***

These parameters should be employed for the BWR recriticality analysis with the RECRIT code in this NKS Project. It should be noted, that previous recriticality analysis performed by Anttila (1990) showed that a serious recriticality threat in a BWR core without control rods emerges only when the intact core is either filled with a water-steam mixture of 60% void fraction to a height of 1 meter or filled with saturated water to a height of 0.4 m. Thus, we do not expect that for the case of the non-energetic interaction of the fragmented control rod melt of ~1200 Kg, there should be a reactivity induced accident (RIA) due to recriticality of the core. Of course, this has to be confirmed by the calculations performed with the RECRIT code.

6.2. Task 2: Control Rod Melt Drop into Lower Head with the Occurrence of a Steam Explosion

In this section we will postulate that the control rod melt which is primarily steel, would undergo energetic MCI. It should be recognised again that the initial conditions for the control rod melt are not conducive to an energetic steam explosion due to very low superheat of the melt. Nevertheless we will postulate that a steam explosion would occur.

We will deal with 200 kg/s of metallic melt delivered to the water pool. The physical picture of premixing was analyzed in section 6.1. In this section, we consider a steam explosion in the lower plenum. To maximize the impact of a steam explosion, we assume that the explosion was triggered after the first melt already reached the water pool bottom (e.g., triggered by coolant entrapped in the melt). This way, a substantial melt amount has accumulated in the premixture. With the initial jet velocity of the order of 5 m/s, and considering the reduction in the melt velocity in water, the melt reaches the pool bottom within 2 s, leaving about 400 kg of melt in the premixture. Since the water is saturated, a highly voided premixture is expected, which reduces the explosion energetics very significantly. In fact, taking a highly conservative conversion ratio of 15% of a thermal energy of 320 MJ (0.8MJ/kg molten steel relative to water saturation temperature), the mechanical energy of 48 MJ could be released in the explosion.

Due to the explosion venting, no significant water amount would be pushed upwards by the explosion at the very location of the premixture. Instead, the explosion may cause the water pool to slosh, pushing the pool coolant to penetrate into the core. Note however that the water is of very low compressibility, so the process is essentially isochoric. More precisely, the coolant pool's volume shrinks due to the high pressure generated in the explosion, which causes the steam in the premixing zone to condense.

A maximum volume that could form a "slug" is estimated to be of the same volume as that of the premixing zone V_{mix}. Our assessment shows that V_{mix} for an in-vessel MCI from a single-jet situation can be conservatively bounded to $V_{mix} = 3\ m^3$. Given the core fuel bundle cross-sectional area of 7.2 m^2 and the lower plenum area of 15 m^2, about half of V_{mix} (i.e.1.5 m^3) may be pushed into the core through un-blocked pathways (in the fuel channels). The time scale for shock wave propagation and collapse of the premixing zone (L~ 3 m, D~ 1m) is $L/c_{mixture}$ = 3 m/150 (m/s) = 7..20 ms. For the cross-sectional area of about 4 m^2 (2/3 volume is coolant and 1/3 are filled by the fuel elements) the slug penetrates into the core with a velocity of 20..60 m/s during a period of 7..20 ms, raising the water level to about 0.4 m. This volume will stay in the core even when the pool dynamics would favor liquid re-collection into the lower plenum. The reason is that the liquid receding now is resisted by evaporation, and not anymore driven by the pressure wave as during the steam explosion that formed the slug in the first place.

Given the CHF in the hot (previously unwetted) core region is 1 MW/m^2, the linear heat flux on the fuel element is 35 kW/m, or 14 kW for the 0.4 m length. For each fuel element, the amount of added water is 36 g, which would evaporate within 5 s under the 14 kW heat input.

It should be noted that during rapid quenching and for a short time period, a higher than $1 MW/m^2$ heat removal rate is possible, which would further increase the evaporation rate. More importantly, a rapid evaporation may cause rapid/explosive pressurization that could mechanically destruct the lower core region. This process however is much delayed than the recriticality process and hence has no direct influence on the recriticality analysis.

Another equally-probable scenario of slug penetration is when the pool sloshing is asymmetric and the slug penetrates only a half of the core's cross-sectional area. The slug penetration parameters that can be used as coolant conditions for the recriticality analysis are listed below. Uncertainties in steam explosion assessment in complex geometries of the core and lower plenum necessitate a large range for these parameters.

Slug Penetration in the whole core cross-sectional area (rapid and slow penetrations)

Velocity of slug penetration: 60 m/s (20 m/s)
Period of the slug penetration: 7 ms (20 ms)
Void fraction of the slug: 0%
Total coolant volume in core: 1.5 m³
Reflooding height: 0.8 m

Slug Penetration in only ½ of core cross-sectional area

Velocity of slug penetration: 20...80 m/s
Period of the slug penetration: 10 ms
Void fraction of the slug: 0%
Total coolant volume in core: 1.5 m³
Reflooding height: 0.8 m

6.3 Task 3: Evaluation of Scenarios Which May Lead to Recriticality

In this Task we would like to speculate and consider scenarios which might have a potential to provide critical conditions in the core. We will exclude the scenario examined and analysed in the EU's SARA Projects of the cold ECCS water injection to an essentially intact core, except for the melt-down of the control rods. We think that the SARA Project scenario is perhaps the most credible for recriticality, since (a) at that point in time during the severe accident, the core will be in its most critical state if water is added and (b) the ECCS water could be added as a core spray or through the downcomer and (c) the ECCS water is highly subcooled.

We have seen in Sections 6.1 that assuming that all the control rod melt drops into water, and that 75% of its heat content is delivered to the water, does not provide a "sufficiently large level swell", which might lead to core recriticality, although this has yet to be confirmed with the calculations with the RECRIT code, currently being performed by J. Miettinen at VTT. Also in Section 6.2, even though the control rod-melt does not have sufficient superheat, we assumed that the melt would fragment, form a premixture, which leads to a steam explosion. The effect of the resulting addition of a two phase mixture to core, on the core criticality will be calculated with the RECRIT code. The assumptions made in Section 6.2, however, are highly unrealistic and the potential for obtaining a steam explosion and the resulting two phase mixture injection into the core is of very low probability. The control rod melt is not large or hot enough to pose a serious threat for steam explosion.

We mentioned in Section 1 (Background) that there may be a time window during the progression of the core melting in its original configuration, in which there is a greater potential for a larger level swell and a more energetic steam explosion. Clearly, the core heat-up is proceeding further after the melt-down of the control rods, and later in time, the Zircaloy clad the fuel pellets and the Zircaloy shroud of the fuel bundle will melt and accumulate on the core plate. The core plates heats up further and it is possible that it may fail and a larger quantity of melt may drop into the water contained in the lower head. However, it should be noted that the corium exit of the fuel and the Zircaloy from the core reduces its criticality considerably. There is a greater potential for a large swelling of the lower plenum water and a large probability of a steam explosion and the resulting addition of a water slug into the core region. However, it should be noted that the corium exit of the fuel and the Zircaloy from the core reduces its criticality considerably.

This is essentially what was considered in the original STUK study (Okkonen et al. 1993). That study assumed that 4-8 tonnes of melt in a very well fragmented state (1 mm size particles) would deliver from 200-25000 MW for production of steam during an energy release and transfer period of 0.5 to 12 seconds. The resulting slug filled the core with water within a second at ~600 Kg/m^3. This, coupled with the assumption an intact core, denuded of control rods, of course, produced a reactivity spike.

The Okkonen analysis made very conservative assumptions, on the melt delivery to the water in the lower head, its fragmentation and the resulting steam production rate. Certainly, with the steam production, which during its upward flow will deter the downward flow of melt, it would take much much longer to deliver the melt to the water than the maximum of 12 seconds assumed. The fragmentation and the particle size assumed are also contrary to the FARO data, which shows larger size particles and formations of a cake which settles at the bottom without transferring much of its heat to the water. Thus, the assumptions made in arriving at the water delivery rate to the core in the STUK study are very unrealistic and the quantity and rate of water addition would be much much smaller.

Another assumption made in the STUK study is that the core has all its fuel bundles intact and in place, and that the control rods are absent is also highly unrealistic for the calculation in which a large quantity of corium is added to water. Moving the UO_2 from its most critical location in the core into a melt pool at the core plate reduces the core reactivity substantially. In a relatively radially-flat power BWR core, the UO_2 melting in the interior 80% of the core would be quite coherent and the assumption of a relatively intact core with injection of several tonnes of corium melt into the lower head is inconsistent. We believe that it is inconsistent to assume that the core is in a sufficiently reactive state to achieve recriticality after the UO_2 melting process has progressed.

The injection of corium melt into the water could lead to a steam explosion. In particular, if a substantial amount of unoxidized Zr is present in the corium melt, the fragmentation could lead to finer particles. Although no energetics have been observed in the FARO tests, the recently TRIO tests in Korea have shown measurable energetic steam explosion. Thus, the occurrence of steam explosion con not be excluded, however the observed conversion ratios are in the range of 1% or less.

The KROTOS, FARO and the TRIO experiments have all been performed in an open vessel, approximately simulating the geometry of the lower head of a PWR. The BWR lower head, in contrast, contains a forest of control rod guide and instrumentations tubes (see Fig. 5). The control rod guide tubes are located ~300 mm apart and in between two control rod guide tube there is generally an instrumentation tube. Thus, there is not much space available for a large volume premixture formation. The premixtures formed could lead to small-scale steam explosions and one can envision a few small explosions, each having relatively small energy release.

Another issue in the core melt scenario is that of the accessability of any water or two phase swell or slug formed in the lower head to the interior of the molten or damaged core in its original confines. It is most probable that the core melt would have formed blockages or thick crust at its lower boundary and may not allow the entry into core of the water swell or slug coming from the lower head. This would be the case in both of the core melt relocation scenario, i.e. (1) the core melt enters the lower head from somewhere in the middle of the core plate or (2) it enters the lower head from the side as it did in the TMI-2 core melt-down accident- The inability of water or two phase flow mixture to be in the vicinity of the core melt to provide the required moderation of neutrons will preclude core recriticality.

The inability of corium melt and water to co-exist without making steam, which is a very poor moderator, and the inability of maintaining close contact of small quantities of corium(UO_2) with water as designed in the original geometry of the core, leads us to conclude that achieving recriticality after substantial melting of the core fuel bundles is highly unlikely. We believe that the time window for recriticality is when the core fuel rods are in their original state, possibly hot, but not molten and that the control rods are gone. This is the state of the core considered in Tasks 1 and 2. In that state, the injection of water into the core due to level swell or steam explosion, caused by the relocation of control rod melt into the lower head may lead to recriticality. Even in this state, a coherent reflood will be difficult to achieve and the reflood, producing much steam will engulf the core in steam, which is not a very effective neutron moderation agent. After that time window, when the fuel melting starts and corium melt is available for relocation, larger level swells or steam explosion-induced water slugs will either not have access to the core due to the blockages at its bottom boundary, or if an access can be found, the mixing of water and corium melt will lead to corium fragmentation and dispersal or the formation of a corium melt zone surrounded by water, which is not an effective geometry for recriticality. The TMI-2 accident has provided this evidence.

We believe that an analysis of the recriticality potential should include a determination of the core subcriticality in its damaged state, as predicted by a code like MELCOR. It should consider the core configuration after UO_2 melting has started and as the core geometry changes and as corium relocation occurs. The subcriticality caused by these changes should be estimated by a neutronics code. Also, the amount of water or two phase mixture needed to bring the core back to recriticality should be calculated. This will provide a reasonable estimate if time windows exist, after fuel pin damage and UO_2 relocation has started when it may be possible to bring the core back to critical. Such an analysis will complement the analysis performed by Antilla for a core in its original configuration, except for the loss of control rods.

7. Conclusions and Recommendations

The study described in the preceeding pages comes to the conclusion that the most likely time window for a recriticality of the BWR core during a postulated core melt accident is when the control rod melt is deposited on the core bottom plate and the core fuel bundles are intact and in place. The recriticality can occur by addition of water or a two phase mixture to core by either (a) ECCS injection of cold unborated water or (b) a level swell or a slug of water injected from the lower head when the control rod melt drops into the water of the lower head.

The potential of recriticality due to ECCS injection has been investigated in the EU's SARA Project, while the analysis conducted in this study is concerned with the core reflood characteristics due to interaction of the control rod melt with the lower head water.

The reflood characteristics were derived for the cases of (i) non energetic and (ii) energetic MCI. It was found that the core reflood due to non-energetic interactions would most probably not lead to recriticality, while that due to energetic interactions might. Both of these conclusions have to be confirmed by calculations performed with the RECRIT code.

Other scenarios for recriticality due to water addition from the lower head resulting from subsequent MCIs were examined. Larger additions of water (or two phase mixtures) into the core without control rods are feasible later in the core-melt scenarios, when large amounts of the very high temperature corium melt may drop into the lower head. It is, however, argued that with the redistribution of the fuel in the core and/or the drop of fuel into the lower head, the core would become highly subcritical and may not become critical by addition of limited quantities of water or two phase mixture. There are additional limiting factors, e.g. (i) the difficulty in access of the water or two phase mixture to the core due to blockages at the core plate, or higher in the cores and (ii) the difficulty in obtaining the appropriate water/UO_2 atom ratios and geometry for criticality. The most probable configuration would be that of a particulate bed or a corium pool surrounded by water, which are not efficient geometries for criticality. We believe achieving recriticality is highly unlikely after core has melted and melt has collected at core plate.

It is recommended that further work on this topic should include (a) further definition of the scenario, (b) appropriate description of the core blockages and the melt drop scenario, (c) appropriate description of the melt water interaction process in the lower head of the BWR and (d) analysis of core subcriticality due to the core melting process.

The item (c) should be described mechanistically providing quantification of (i) premixture volume, while accounting for the forest of the control rod guide tubes in the lower head, (ii) explosion or non-explosion induced pool sloshing and the liquid push-away velocity at the free surface level and (iii) the actual level swell or slug penetration into the core while taking into account the core plate, fuel element and blockage configurations.

The item (d) should include neutronic analysis of the core subcriticality consistent with the configuration of the core, i.e. the core geometrical configuration accounting for the movement of the control rod material and of fuel, consistent with that calculated from a code like MELCOR. This analysis should provide the magnitude of the water addition to the core which would make the core critical.

References

Anttila M., "Recriticality Potential in a Severe Accident at TVO plant", in Finnish, VTT YDI013/90 (1990). Quoted from Okkonen et al (1993)

Bandurski, Th., Cabezudo, C., Methews, D., Knoglinger, E., “Recriticality of a BWR core during reflood after control blade meltdown” ANS Winter Meeting ’94, Washington, D.C., November 13-17, (1994)

Frid W. et al, “Severe Accident Recriticality Analysis, (SARA)”, EU Project Fourth Framework Program EC-INV-SARA (99) D016 SKI Report 99:32 (Nov.1999)

Hofmann P., Markiewicz M.E., and Spino J.L., “Reaction Behavior of B_4C Absorber Material with Stainless Steel and Zircaloy in Severe LWR Accidents”, Nuclear Technology, V.90, 226-244 (1990)

Lindholm I.,”Calculations of Core Status Prior to Anticipated Refloding/Recriticality with MELCOR 1.8.3”, VTT ROIMA-7/97 (1997)

Magallon, D., Huhtiniemi, I., “Corium Melt Quenching Tests at Low Pressure and Subcooled Water in FARO”, Nuclear Engineering and Design, 204, (2001)p. 369-376

Mosteller, R.D. and Rahn, F.J., “Monte Carlo calculations for recriticality during the reflood phase of a severe accident in a BWR” Trans.Am.Nucl.Soc., 63, 254. (1991)

Okkonen T.J., J. Hyvarinen, and K. Haule, “Safety Issues Related to Fuel-Coolant Interactions in BWRs”, Proceedings of OECD/CSNI Specialist Meeting on FCI, Santa Barbara, pp. 296-308 (1993)

Sairanen R., “Recriticality due to FCI in the RPV Lower Head: Initial Conditions”, VTT Energy-Nuclear Energy, August 2001, ENE4-PR-6/01 (2001)

Shamoun, B.I. and Wirr, R.J. “Parametric study of reciriticality in a boiling water reactor severe accident”, Nuclear Technology, Vol. 107, (1994)

J.H. Song, I.K. Park, Y.J. Chang, Y.S. Shin, J.H. Kim, B.T. Min, S.W. Hong, H.D. Kim, "Experiments on the interactions of molten ZrO2 with water," Nuclear Engineering and Design 213 (2002) 97–110.

Theofanous TG, Yuen WW, Angelini S, Sienicki JJ, Freeman K, Chen X, Salmassi T, "Lower head integrity under steam explosion loads", Nuclear Engineering and Design, 189 (1-3): 7-57 May 1999.

Bibliographic Data Sheet **NKS-84**

Title	The possibility and the effects of a steam explosion in the BWR lower head on recriticality of a BWR core
Author(s)	B.R. Sehgal & T.N. Dinh
Affiliation(s)	Sehgal Consult, Sweden
ISBN	87-7893-142-8
Date	December 2002
Project	NKS/SOS-2.3
No. of pages	27
No. of tables	3
No. of illustrations	7
No. of references	12
Abstract	The report describes an analysis considering a BWR postulated severe accident scenario during which the late vessel automatic depressurization brings the water below the level of the bottom core plate. The subsequent lack of ECCS leads to core heat up during which the control rods melt and the melt deposits on the core plate. At that point of time in the scenario, the core fuel bundles are still intact and the Zircaloy clad oxidation is about to start. The objective of the study is to provide the conditions of reflood into the hot core due to the level swell or a slug delivered from the lower head as the control rod melt drops into the water. These conditions are employed in the neutronic analysis with the RECRIT code to determine if the core recriticality may be achieved.
Key words	Severe accidents, re-criticality, melt-coolant interaction

Available on request from the NKS Secretariat, P.O.Box 30, DK-4000 Roskilde, Denmark.
Phone (+45) 4677 4045, fax (+45) 4677 4046, e-mail nks@catscience.dk, www.nks.org.