Vitrification In Iran

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Joint ICTP-IAEA International School on Nuclear Waste Actinide Immobilization

IRAN


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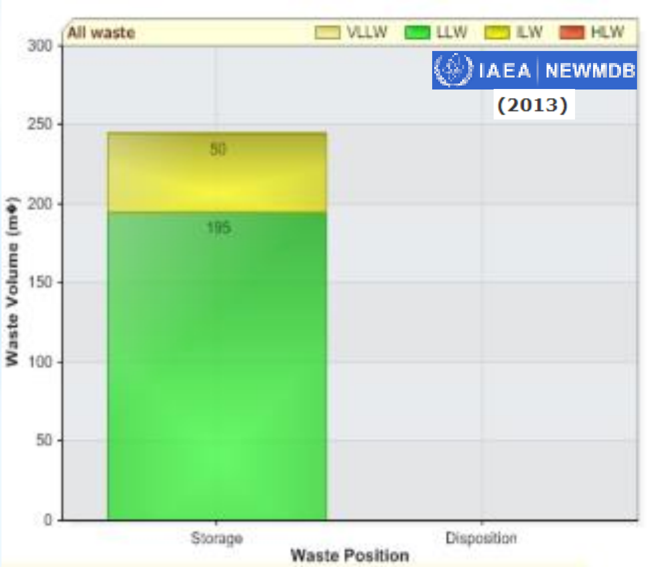
Abu Dhabi
U.A.E.
Muscat
Mascat
SAUDI ARABIA
OMAN

Potential Sources of Nuclear Waste in Iran:

- **Nuclear Operation:**
- A large nuclear power reactor (VVER-1000- 915 MWe net),
- the Tehran Research Reactor (TRR)-5 MW pool-type research reactor,
- **Fuel Cycle operation:**
- The mining and milling (Gachin/Gchine, Saghand),
- The Bandar Abbas Uranium Production Plant (BUPP) ,
- Ardakan Uranium Production Plant,
- A uranium conversion plant (UCF) at the Isfahan
- Enrichment plant: Iran is now limiting,
- A fuel fabrication plant next to the UCF,
- **Nuclear Applications**
- Agricultural, industrial and medical application



Waste Management Strategy In Iran

- Iran Nuclear Waste Management Company is the only authorized company for radioactive waste management in Iran which acts under framework of Iran nuclear regulatory authority and AEOI

- **Main activities:**
- Waste treatment
- Cementation
- Interim storage
- Near surface disposal

- Nuclear Science and Technology research institute do research for establishment, development, promotion and optimization of methods and processes for waste management

HLW in Iran

- **Joint Comprehensive Plan of Action (JCPOA) (** 20 July 2015) **:**

- For 15 years Iran will not, and does not intend to thereafter, engage in any spent fuel reprocessing or construction of a facility capable of spent fuel reprocessing, or reprocessing R&D activities leading to a spent fuel reprocessing capability, ***with the sole exception of separation activities aimed exclusively at the production of medical and industrial radio-isotopes from irradiated enriched uranium targets.***

- Atomic Energy Organization of Iran (**AEOI**) has implemented a program to produce ^{99}Mo as a radiopharmaceutical, by separation from the fission products of irradiated ^{235}U.

- In this process, HLW is produced.

- At this time, there is no plan for conditioning of HLW produced in this process, and the interim storage as a liquid form is the option chosen by Iran Nuclear Waste Management Company .

Vitrification in Iran

- ❑ Immobilization of simulated HLW generated by the radiopharmaceutical production unit in the borosilicate glass
- ❑ Sorption of HLW species onto the aluminumsilicate type adsorbents followed by heat treatment

- Synthesis of novel adsorbents
- Using the natural adsorbents
- Separation of radionuclide from the nuclear wastes by sorption process has the potential of significantly decreasing the costs of the immobilization and disposal of the radioactive waste by minimizing waste volumes

Vitrification of simulated HLW

- This study deals with production of durable borosilicate glasses for the immobilizing of radioisotopes from nuclear waste streams generated by a radiopharmaceutical production unit.
- In this regard, different boron frits (glasses) and waste-loaded glasses were prepared under various experimental conditions.
- The effects of some parameters such as **melting temperature**, **cooling procedure** and **various raw materials** were investigated.
- In order to determine the best waste loaded glass composition, **two different standard leach tests** were performed using **powdered and disk shape** products.
- All experiments were performed in nonactive bench scale.

Ref: Vitrification of HLW generated by a production unit for radiopharmaceuticals using simulated waste solutions. *Journal of radioanalytical and nuclear chemistry*, *261*(3), 619-623

Composition of the frits and simulated wastes

Table 1. Composition of synthesized frits (in %)

Composition	G1	G2	G3	G4	G5	G6	G7	G8
Al_2O_3	2.36	8.24	2.36	9.46	10.6	11.36	9.38	9.37
B_2O_3	11.24	7.95	11.24	7.85	11.6	12.2	10.09	10.09
CaO	3.72	2.59	3.72	2.55	8.57	9.17	7.57	7.56
MgO	1.9	0.96	1.9	0.95	3.74	4.01	3.34	3.35
Na_2O	25.12	11.25	25.12	11.1	11.18	11.98	27.73	27.71
Na_2CO_3	-	-	-	-	27.16	29.13	23.53	23.5
SiO_2	51.41	69	51.41	68.09	16.3	17.5	14.46	14.47
$Fe(NO_3)_3$	-	-	-	-	2.67	2.87	2.38	2.42
TiO_2	5.46	-	5.46	-	8.18	1.78	1.52	1.51

Table 2. Composition of simulated wastes (in %)

Waste code	CsO_2	BaO_2	SrO_2	Y_2O_3	La_2O_3	Nd_2O_3	Ce_2O_3	ZrO_2	TeO_2	MoO_3
1	13.5	5.35	4.75	2.65	5.1	18.5	10.5	19.31	2.05	18.1
2	16.73	6.63	5.88	3.28	6.32	22.92	13.01	-	2.54	22.43

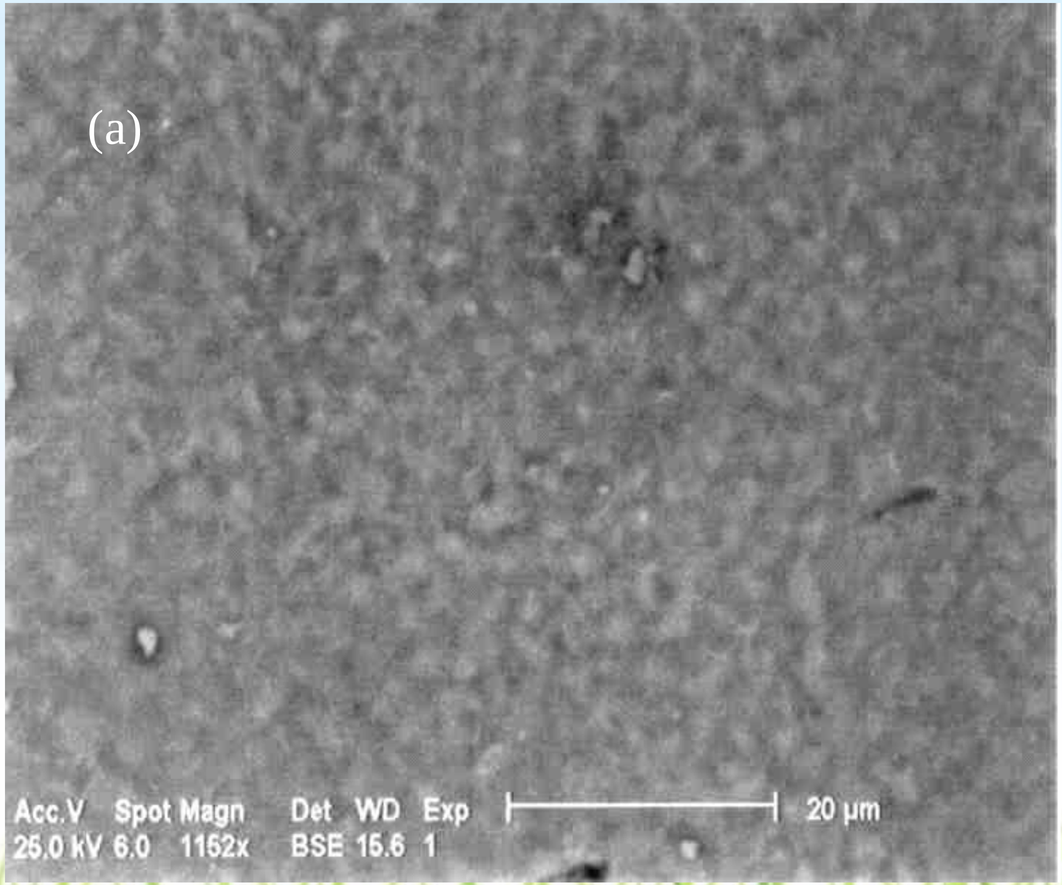
Ref: Vitrification of HLW generated by a production unit for radiopharmaceuticals using simulated waste solutions. *Journal of radioanalytical and nuclear chemistry, 261*(3), 619-623

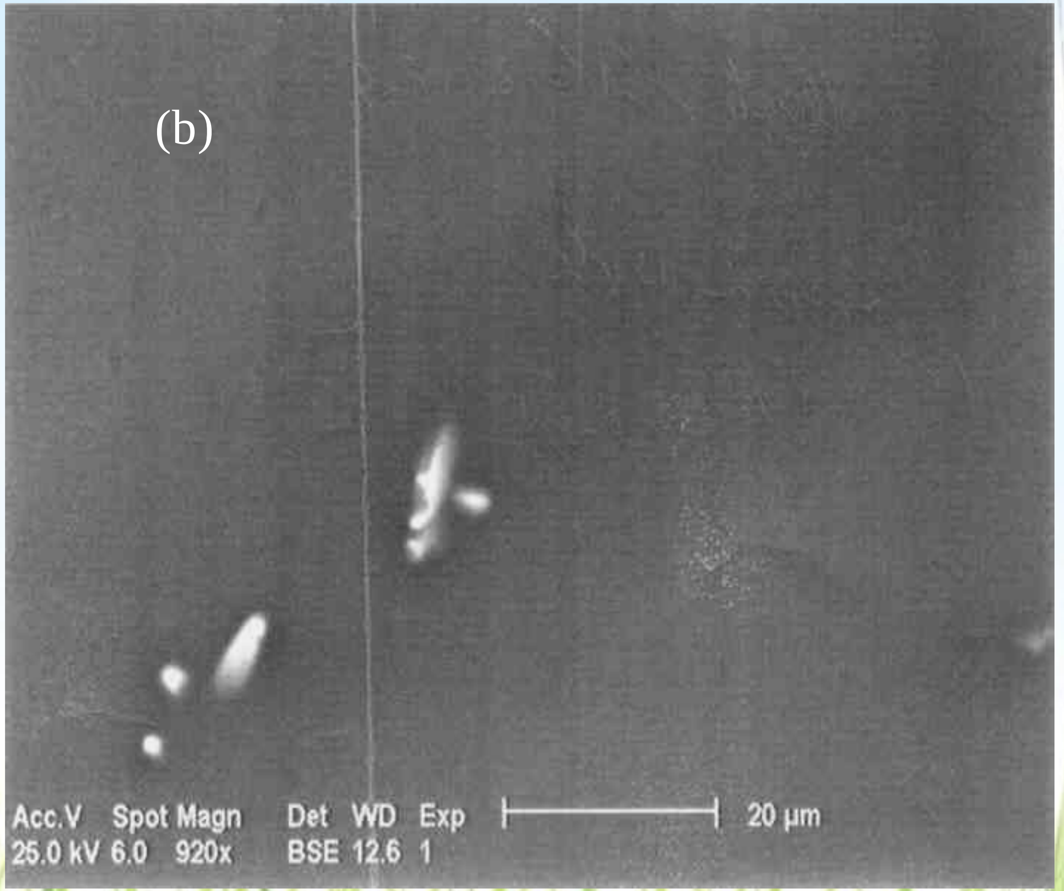
Optimized operating conditions

Operating conditions	Heating operation	Melting point, °C	Vessel type	Cooling procedure	Method of homogenization
G6	25 °C (with a rate 500 °C/h) … 500 °C (0.5 h remained) (500 °C/h) …. 1200 °C (1.6 h remained)End.	1200	Crucibles of China	Pouring in distillated water	Wet method
GW7	25 °C (with a rate 300 °C/h) …. 500 °C (0.5 h remained) (300 °C/h) …. 1200 °C (2 h remained)End.	1200	Crucibles of ceramic	Pouring on the stainless steel sheet	Wet method

Ref: Vitrification of HLW generated by a production unit for radiopharmaceuticals using simulated waste solutions. *Journal of radioanalytical and nuclear chemistry*, *261*(3), 619-623

SEM images of (a)G6 frit and (b)GW7 waste-loaded glass





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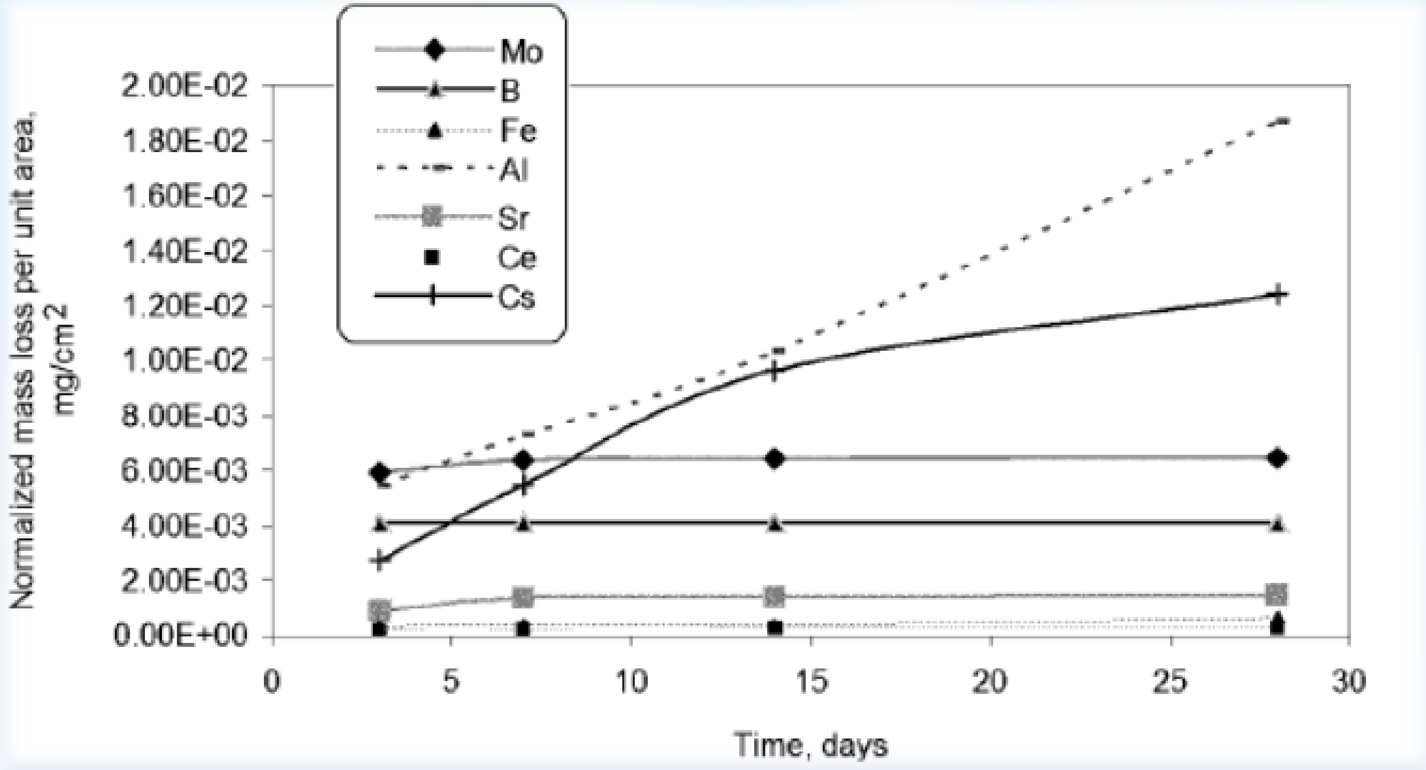
Leach tests of disk shape waste form



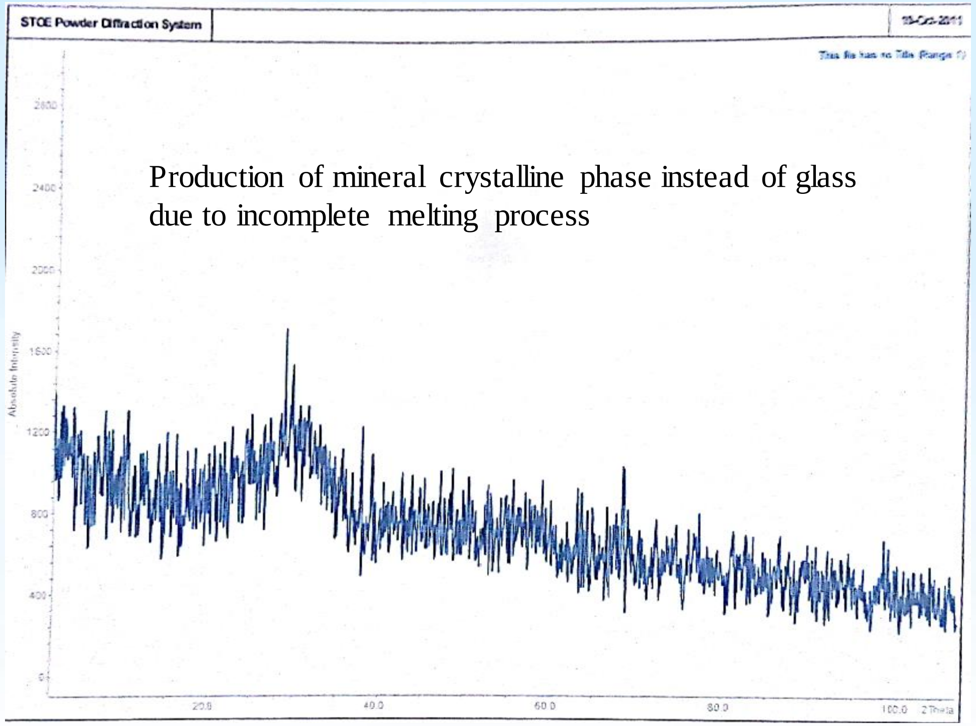
According to the results obtained from leach resistance tests, stability and structural investigations using SEM micrographs, the GW7 sample with a **glass to waste ratio of 85 : 15** was the most suitable matrix

Ref: Vitrification of HLW generated by a production unit for radiopharmaceuticals using simulated waste solutions. *Journal of radioanalytical and nuclear chemistry*, *261*(3), 619-623

Making borosilicate glasses for Cs and Sr immobilization

Composition of synthesized glass

Composition	Weight Percentage (%)
Al_2O_3	9.45
B_2O_3	11.81
CaO	7.87
MgO	3.94
Fe_2O_3	7.87
Na_2CO_3	31.50
SiO_2	23.62
$SrCl_2$	1.57
$CsNO_3$	1.57
Li_2CO_3	0.80

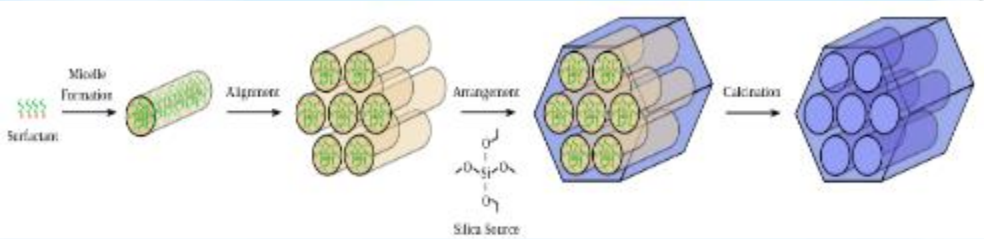


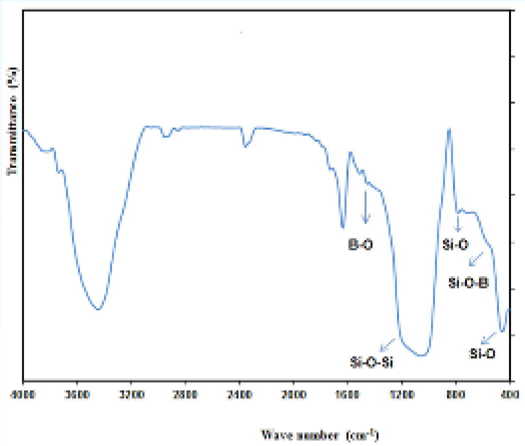
Production of mineral crystalline phase instead of glass due to incomplete melting process

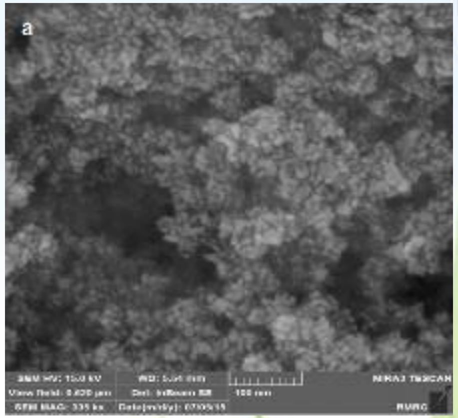
The study of acidic and alkaline type of environments on Cesium and Strontium leaching showed that the Cs leaching rate is acceptable, but the Sr in this environment is significantly being leached.

Mesoporous borosilicate and investigation of its performance for adsorption and immobilization of Cs and Sr

- mesoporous materials have found great utility as sorption media because of their large internal surface area and more adsorption sites than other adsorbents, which caused the increasing attention of researchers to use them in the nuclear waste management.
- In this study, nanoporous alumioborosilcate (Al-B-MCM-41) was prepared as new adsorbent and was used
- MCM-41 (Mobil Composition of Matter No. 41) is a mesopores material with a hierarchical structure from a family of silicate and alumosilicate solids that were first developed by researchers at Mobil Oil Corporation

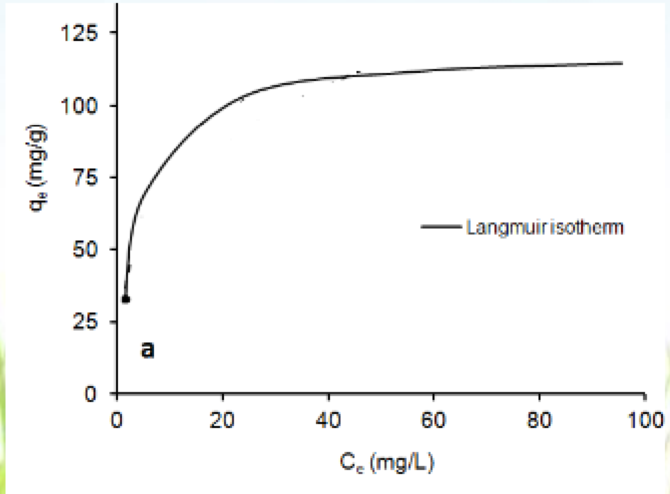






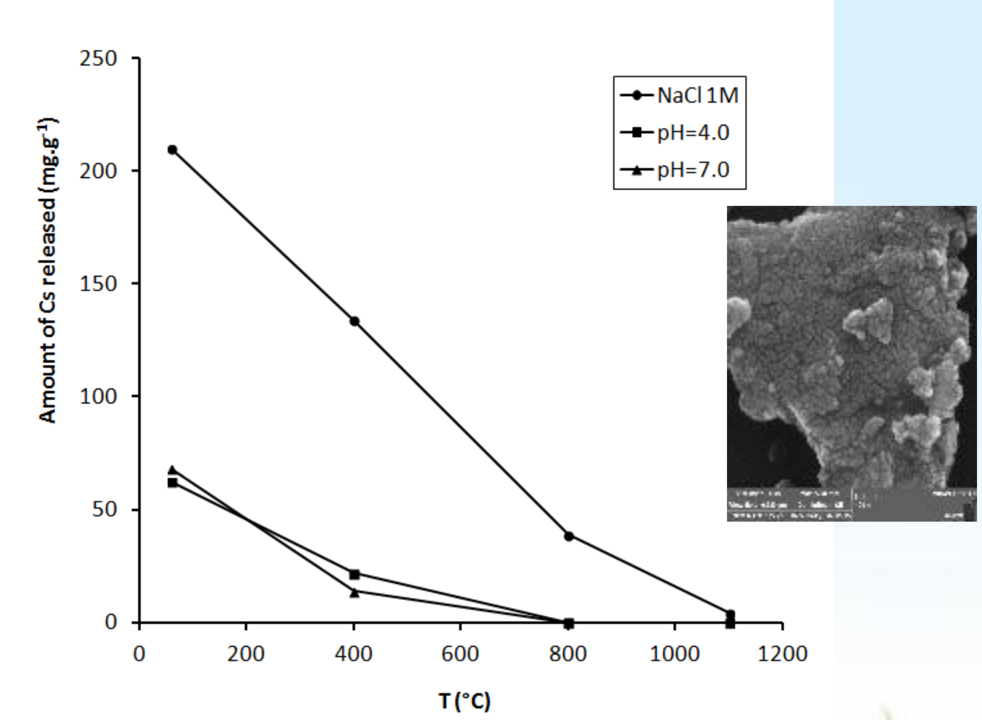
Adsorption Results

- The effects of various parameters like the **initial pH value** of the solution, **contact time**, **temperature**, **ionic strength of solution**, **interference ions** and **the initial concentration of the metal ions** (strontium and cesium) on the adsorption efficiencies has been studied systematically by batch experiments.
- The results show that maximum adsorption capacity of cesium and strontium onto nanoporous alumioborosilcate (Al-B-MCM-41) were found 119.05 and 125.00 $mg.g^{-1}$, respectively.



Leaching Results

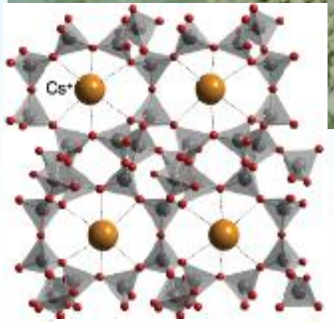
- Cesium and strontium adsorbed aluminoborosilicate were heated at different temperatures, and the heat-treated materials with leaching test were investigated.
- Leaching resuts show that immobilization ability of Cs and Sr ions in the heat-treated materials increased as the treatment temperature were increased.



Pellet is produced using a hydraulic press and a stainless steel extruder at 400 $g.cm^{-2}$ load pressure. Leaching test time is 24 h

Adsorption and immobilization of Cs radionuclide on the clinoptillolite

- Zeolites due to their high thermal and radiation stability, selectivity and high exchange capacity are considered for removal of Cs radionuclides from aqueous solution.
- The natural zeolites of sabzevar area (clinoptillolite) was employed for Cs adsorption and immobilization.
- The effects of various parameters like the **initial pH** and **ionic strength of the solution**, **contact time**, **temperature, interference ions and the initial concentration of the cesium** on the adsorption efficiencies of clinoptilollite were studied.
- The maximum Cs adsorption capacity of clinoptilollite were found **172.4 mg.g^{-1}**.



Clinoptilolite is a natural zeolite comprising a microporous arrangement of silica and alumina tetrahedra. It has the complex formula:
$(Na, K, Ca)_{2-3}Al_3(Al,Si)_2Si_{13}O_{36}\cdot 12H_2O$

Heat treatment of Cs loaded Clinoptilolite

- The Cs loaded clinoptilollite were heat-treated at different temperatures and the possibility of Cs immobilization was investigated using leaching tests.
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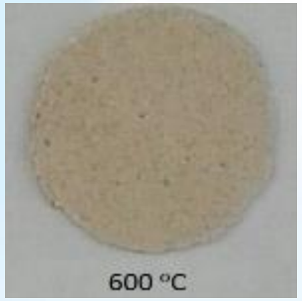
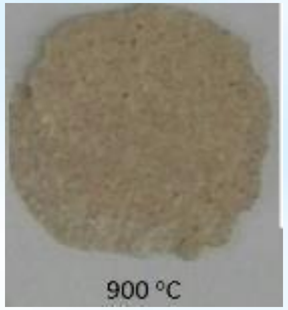
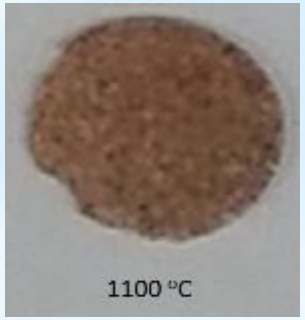
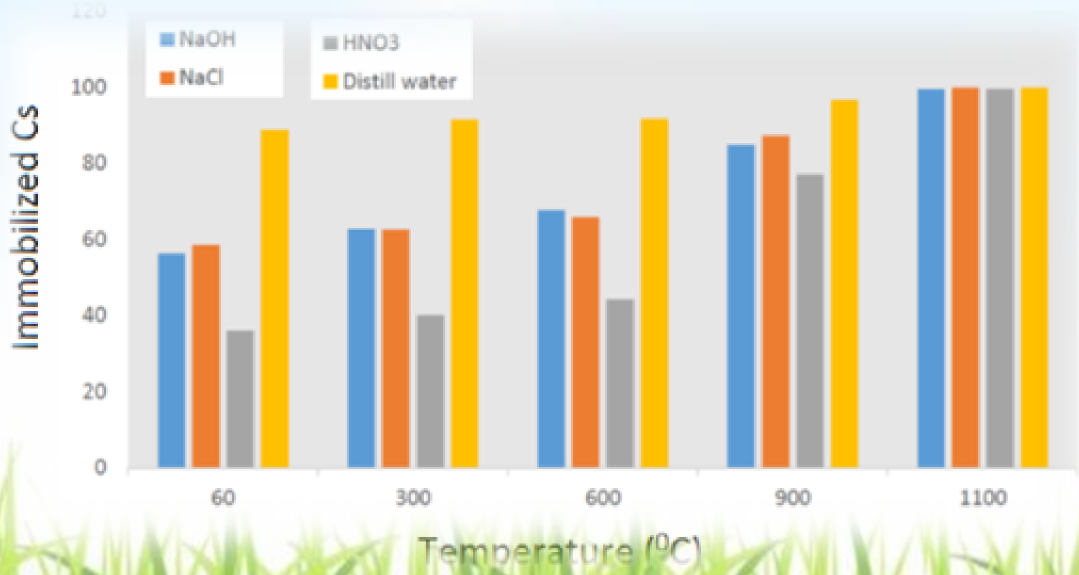








Vitrification of Cs and Sr loaded zeolites

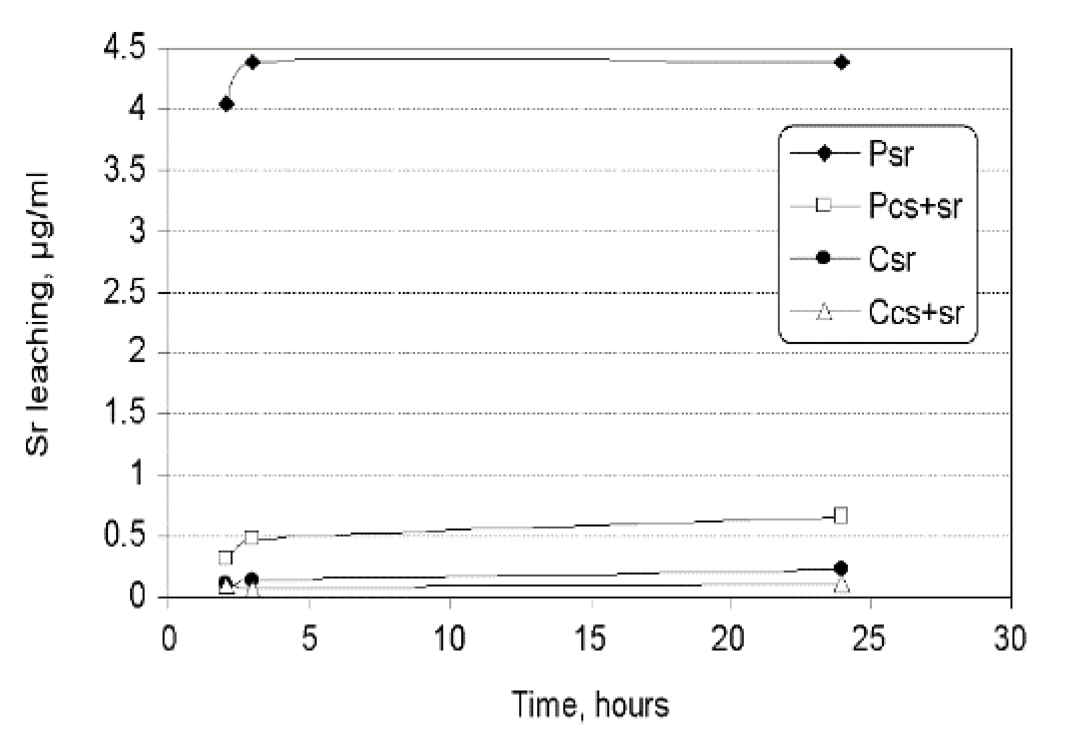
- Cs^{+} and Sr^{2+} adsorbed by natural clinoptilolite from Ardakan region (Yazd province) of Iran and its relevant zeolite P
- Vitrification process of the Cs and Sr Loaded zeolite as a **borosilicate glass** was investigated.

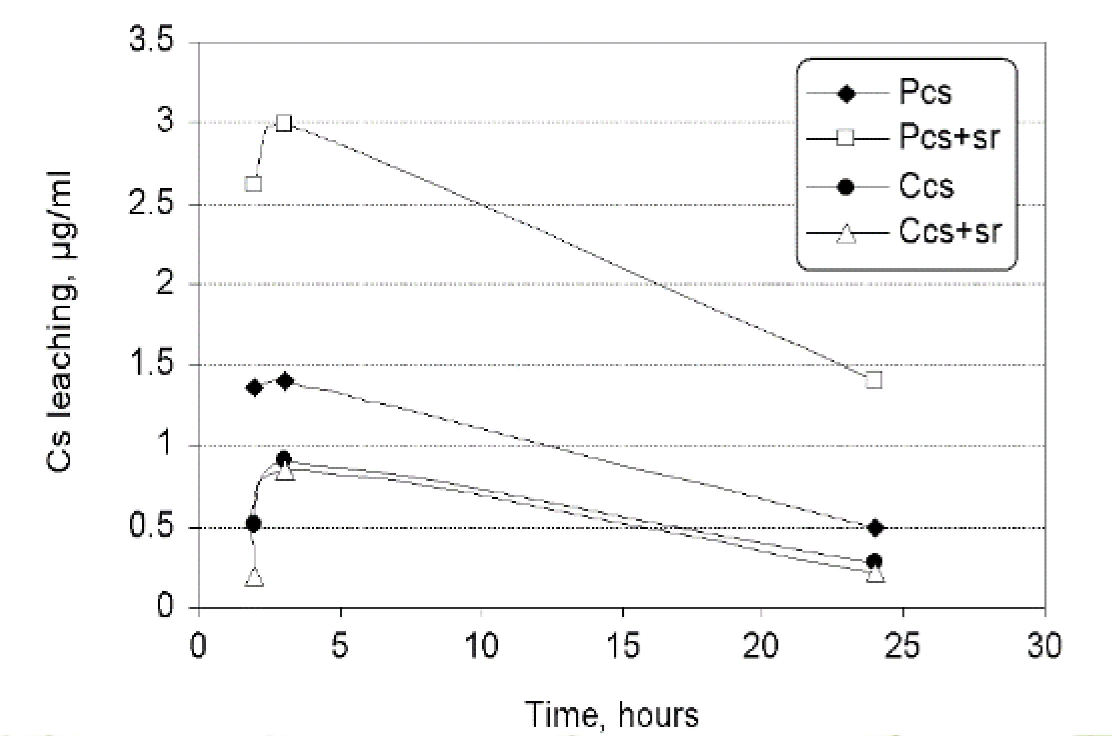
Component	SiO_2	Al_2O_3	Na_2O	K_2O	CaO	MgO	TiO_2	Fe_2O_3	SrO	BaO	L.O.I.*
Clinoptilolite	69.50	14.38	1.23	1.59	4.50	2.33	0.38	2.01	0.40	0.26	6.00
Zeolite P	66.4	19.4	2.2	1.7	2.96	2.7	0.5	3.1	0.4	0.3	6.00

Composition	Al_2O_3	B_2O_3	CaO	MgO	$Fe(NO_3)_3$	Na_2O	SiO_2	**Na_2CO_3**	**TiO_2**
Wt (%)	11.36	12.2	9.17	4.01	2.87	11.98	17.5	29.13	1.75

Ref: Vitrification of Cs and Sr load(2004). ed Iranian natural and synthetic zeolites. *Journal of radioanalytical and nuclear chemistry*, *267*(1), 219-223 (2005)

Leaching Rate





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