SUOMEN ATOMITEKNILLINEN SEURA-ATOMTEKNISKA SÄLLSKAPET I FINLAND

# ATS Tiedotuslehti n:o 3/1973

Sisältö:

VALTION TEKNILLINEN TUTKIMUSKESKUS
Otaniemi 1973-06-11

Suomen Atomiteknillisen Seuran tutustumismatkalle Ruotsin ja Tanskan ydintutkimus- ja ydinvoimalaitoksiin 1973-05-08...13 osallistuneiden VTT:n tutkijoiden matkakertomus

## 1. YLEISTÄ

Suomen Atomiteknillinen Seura järjesti toukokuun 8...13 päivänä 1973 liitteenä 1 olevan ohjelman mukaisen koulutus- ja tutustumismatkan Ruotsin ja Tanskan ydinalan laitoksiin. Matkan aikana tutustuttiin AB Atomenergin tutkimuskeskukseen Studsvikissä, Oskarshamnin ydinvoimalaitokseen, Uddcombin tehtaaseen, Barsebäckin ydinvoimalaitostyömaahan ja Risön tutkimuskeskukseen.

Matkan osanottajaluettelo on liitteenä 2. VTT:stä matkalle osallistui Kauppa- ja teollisuusministeriön rahoituksella tekn.lis. Markku Rajamäki ja dipl.ins. Eero Patrakka Reaktorianalyysiryhmästä, diplomi-insinöörit Seppo Salmenhaara ja Juhani Johansson Reaktorilaboratoriosta, dipl.ins. Seppo Vuori Lämmitysreaktoriryhmästä, dipl.ins. Markku Winter Dynamiikkaryhmästä ja diplomi-insinöörit Jukka Kangas ja Jorma Väkiparta Luotettavuusryhmästä. Lisäksi matkalle osallistui VTT:sta omalla kustannuksellaan dipl.ins. Frej Wasastjerna Reaktorianalyysiryhmästä, dipl.ins. Antero Tiitta Reaktorilaboratoriosta, dipl.ins. Risto Tarjanne Lämmitysreaktoriryhmästä ja tekn.lis. Juhani Ervamaa Luotettavuusryhmästä.

Seuraavassa on selostettu matkakohteittain laitoksia, joihin tutustuttiin, ja matkalla saatuja vaikutelmia.

## 2. STUDVIKIN TUTKIMUSKESKUS

Opintomatkan ensimmäinen tutustumiskohde oli AB Atomenergin atomiteknillinen tutkimuskeskus Studsvikissä. Tutustumisen aluksi pidettiin luentosalissa n. tunnin kestänyt yleisesittelytilaisuus, jossa luotiin yleiskatsaus Studsvikissä suoritettavaan tutkimustoimintaan sekä sähköenergian tarpeeseen Ruotsissa ja sen tyydyttämistä varten kaavailtuun reaktorien rakennusohjelmaan. Lisäksi tässä tilaisuudessa esiteltiin esijännitetystä betonista valmistettavan reaktoripaineastian kehitysprojektin tämän hetkinen tilanne. Betonipaineastialla katsottiin olevan useita etuja verrattuna tavanomaiseen teräspaineastiaan mm. suurempi turvallisuus ja että suurikin yksikkökoko voidaan toteuttaa ilman kuljetusongelmia. Toisaalta idean soveltamista käytäntöön hidastaa puute käytännön kokemuksista.

Aamupäivän ensimmäisenä tutustumiskohteena olikin mittakaavassa 1:3,5 tehty betonipaineastia, joka on rakennettu v. 1969. Tällä mallilla on suoritettu erilaisia lämpötila- ja ylipainetestauksia sekä lisäksi onnettomuustilanteita simuloivia kokeita. Kiehutusvesireaktorin osalta on jo tällä hetkellä valmiit suunnitelmat kaupallista soveltamista silmälläpitäen ja tehdyt suunnitelmat ovat suurelta osin sovellettavissa myös painevesireaktoreihin.

Toisena kohteena aamupäivällä oli lämpölaboratorio, jossa tutkitaan mm. lämpö- ja virtaustekniikkaa, hydrauliikkaa ja erilaisia materiaaliongelmia sekä suoritetaan erilaisten laitteiden toimintakokeita. Laboratoriossa on kaikkiaan 15 suurempaa koelaitteistoa eli piiriä, joilla em. tutkimusaloihin liittyvät kokeet suoritetaan.

Lounaan yhteydessä selostettiin Studsvikissä suoritettavaa polttoaineen testaustoimintaa. R2-tutkimusreaktorissa polttoainesauvanäytettä säteilytetään erilaisissa olosuhteissa ja tulokseksi saadaan tietoja polttoaineen geometrian ja lujuusominaisuuksien muuttumisesta palaman ja vallinneen säteilytystason funktiona.

Iltapäivällä ensimmäisenä tutustumiskohteena oli tutkimusreaktori R2, jonka teho on 50 MW(t) ja jäähdytteenä/moderaattorina toimii kevyt vesi sekä heijastimena Be ja $D_2O$. Reaktoria voidaan käyttää erilaisiin säteilytystarkoituksiin ja lisäksi on 6 kiinteätä testauspiiriä, joiden avulla voidaan testata erityyppisten reaktoreiden polttoainemateriaaleja olosuhteissa, jotka lämpötilan, paineen ja neutronivuon suhteen vastaavat kyseisten reaktoreiden ominaisuuksia. Samassa rakennuksessa toimii myös R2-0 reaktori, joka on allastyyppinen, luonnonkierrolla jäähdytetty ja maksimiteho on 1 MW.

Iltapäivän toisena kohteena oli korroosiolaboratorio, jossa testataan eri reaktorimateriaaleja olosuhteissa,jotka simuloivat reaktorilaitoksissa vallitsevia olosuhteita. Laboratorion laitteistot eivät kuitenkaan ole rajoittuneet pelkästään ydintekniikan materiaalien testaukseen, vaan niitä voidaan suureksi osaksi käyttää myös muihin tarkoituksiin.

Iltapäivällä tutustuttiin lisäksi vielä Studsvikin laajaan ydinteknilliseen kirjastoon.

Käynnin aikan oppaina toimivat seuraavat henkilöt:

| | |
|---|---|
| R Nilson, vVD | Industriella Projekt och Marknadsutveckling |
| | |
| S Ahlqvist | Materialforskningsreaktor R2 |
| J Flinta | Värme- och Strömningsteknik |
| B Johnsson | Mekaniska Konstruktioner |
| P Margen | Program och Marknadsutveckling |
| H Mogard | -" |
| B Pershagen | Reaktorer |
| T Petersén | Personal |
| M de Pourbaix | Korrosion och Reaktorkemi |
| K Saltvedt | Materialforskningsreaktor R2 |
| S Sandström | Externa Relationer |

## 3. OSKARSHAMNIN YDINVOIMALAITOS

Tutustuminen laitokseen alkoi isäntien lyhyellä katsauksella Oskarshamnverketiin sekä yleisölle tarkoitetulla filmiesityksellä, missä erityisesti pyrittiin korostamaan turvallisuusnäkökohtia. Tämän jälkeen katseltiin vierailijoille tarkoitettua näyttelyä, joka kertoo ydinvoiman kehityksestä ja itse Oskarshamnin laitoksesta.

Oskarshamn I on Ruotsin ensimmäinen kaupallinen ydinvoimalaitos. Sen omistaa OKG (Oskarshamnverkets kraftgrupp AB), jonka ovat muodostaneet 9 suurta yksityistä ja kunnallista voimayhtiötä.

Oskarshamnin laitos, jonka ensimmäinen voimalaitosblokki on OI, sijaitsee noin 25 km Oskarshamnin keskustasta pohjoiseen Simpevarpin niemellä.

OI:ssa on ASEA ATOMin valmistama 440 MW(e) nettotehoinen kiehutusvesireaktori. Turpiini on STAL-LAVALin tuotantoa. Paineastia on saksalaisvalmisteinen. OI:n alustava sopimus tehtiin 1965 ja toimitus tapahtui avaimet käteen periaatteella. Laitos käynnistyi 1971.

OII, joka sijaitsee OI:n vieressä on vasta rakenteilla ja sen pitäisi käynnistyä kesällä 1974. Myös OII on ASEA ATOMin kiehutusvesireaktori, mutta teho on OI:n tehoa suurempi nettotehon ollessa 580 MW(e). Turpiinin on valmistanut STAL-LAVAL, nyt BBC:n lisenssillä. Paineastia on ruotsalainen UDDCOMBin tekemä. OII:n toimitus tapahtuu siten, että OKG:llä on päävastuu.

Esittelyosuuden jälkeen tutustuttiin ensin OI:een. OI oli käynnin aikana valitettavasti pysähdyksissä sekundääripuolen vian vuoksi (turpiinirikko),jolloin myös myöhemmäksi suunnitellut huolto ym. toimenpiteet oli siirretty ko. seisokin aikana suoritettaviksi. OI:ssä tutustuttiin ainoastaan valvomohuoneeseen,siihenkin yleisöparvekkeelta.

OII tutustuminen oli melkoisesti valaisevampi. Reaktorirakennuksessa käväistiin katsomassa reaktorihallia. Reaktoriastiaa ei näkynyt, vaan ainoastaan puolipallon muotoinen suojakupu ja polttoaineen säilytysaltaat. Reaktoriastian alapuolella olevassa tilassa käytiin vilkaisemassa pääkiertopumppuja ja säätösauvalaitteistoa. Turpiinihallissa oli turpiini asennusvaiheessa, samoin lauhduttimet ym. Halliin oli sijoitettu myös sekundääripiirin pienoismalli. Lopuksi käytiin vilkaisemassa lauhdutusveden sisäänottopaikkaa.

## 4. UDDCOMBIN TEHDAS

Matkan kolmas kohde oli Karlskronassa meren rannalla sijaitseva Uddcombin tehdas. Sen omistavat Ruotsin valtio, Uddeholms Ab ja amerikkalainen yritys Combustion Engineering (25%). Tehdas on perustettu v. 1969 valmistamaan pääasiallisesti ydinvoimaloiden raskaita paineastioita ja muita komponentteja. Tätä tarkoitusta varten se on varustettu 800 tonnin nosturilla, jolla lastaus laivaan voidaan suorittaa suoraan tehtaalta. Veden syvyys laiturin kohdalla on 6 m. Uddcombin tehdasta mainostetaankin Euroopan kehittyneimpänä reaktoripaineastioiden valmistajana.

Jonkinlaisen käsityksen tehtaan toiminnan laajuudesta saa tarkastelemalla toimitettujen tai valmisteilla olevien suurimpien ydinvoimalakomponenttien luetteloa: 6 kpl reaktoripaineastioita, 3:een reaktoriin reaktoripaineastian sisäosia, 3 kpl paineen pitäviä suojarakennuksia, 3 kpl höyrygeneraattoreita, 3 kpl muita paineastioita, 2:en reaktoriin primaariputkikomponentteja ja 3 kpl PS-kupoleja.

Tehtaan tärkeimmät tuotantolaitokset ovat konepaja, laboratorio, radiografialaitos ja päästöhehkutusuuni. Näistä tutustuimme laboratorioon ja konepajaan. Varsinkin konepaja oli vierailukohteena onnistunut. Siellä oli helppo tarkastella paineastioiden valmistuksessa käytettyjä menetelmiä ja myös useita välivaiheita oli nähtävissä kuten esimerkiksi Barsebäck 1:n, lähes valmis, 520 tonnia painava paineastia ja Ringhals 3:een tuleva, vielä osina oleva, 350 tonnin paineastia. Silmiinpistävimpiä työvaiheita olivat yli 20 cm-paksuisten levyjen taivutus kaariksi, näiden hitsaus sylinterirenkaiksi, paineastian rungon kokoonpano näistä renkaista, paineastian sisäpuolen hitsaamalla tapahtuva päällystäminen ruostumattomalla teräksellä ja säätösauvojen läpivientien kiinnihitsaus. Viimeksimainittu työvaihe on tehtävä käsin ja sen suorittaminen kestää n. 3 vuotta ja muodostaa näin suurimman osan kokonaisvalmistusajasta.

Yleisvaikutelmana tehtaasta voi sanoa, että käytetyt työkoneet ja -laitteet vaikuttivat jotenkin kääpiömäisiltä massiivisten paineastiain osien rinnalla. Tästä syystä voi olettaa tulevaisuudessa tapahtuvan melkoista kehitystä tällä tekniikan alalla. Jo nyt on kokeiltu päällystyshitsauksessa leveämpää kuin käytössä olevaa n. 0,5 cm:n saumaa.

## 5. BARSEBÄCKIN YDINVOIMALAITOSTYÖMAA

Sydsvenska Kratf AB, jonka pääosakkaina ovat Etelä-Ruotsin kaupungit, rakennuttaa Barsebäckiin kahta ydinvoimalayksikköä, jotka kumpikin ovat teholtaan 580 MWe. Reaktorit toimittaa ASEA ATOM, ja ne tulevat olemaan miltei samanlaiset kuin Oskarshamn II:n kiehutusreaktori. Polttoainenippujen lukumäärä on kuitenkin hieman pienempi (444><436) ja nipun kokonaispituus hieman suurempi (4620mm><4385mm).

Samalle paikalle rakennettanee myöhemmin kaksi lisäyksikköä, jotka on suunniteltu otettavaksi käyttöön 1980-luvun puolivälissä.

Barsebäck I:n rakennustyöt olivat edistyneet siihen vaiheeseen, että oli mahdollista ryhtyä reaktoripaineastian asennukseen. Varsinaiset laitteistoasennukset eivät kuitenkaan vielä olleet alkaneet, ja rakennusten valutyöt olivat vielä kesken. Barsebäck II:ssa olivat käynnissä pohjarakennustyöt.

Voimalaitoksen sijaintipaikassa on huomionarvoista se, että se on yhtä kaukana Malmöstä ja Kööpenhaminasta. Niinpä etäisyys Barsebäckistä Juutinrauman vastakkaisella puolella sijaitsevaan Kööpenhaminaan on vain n. 20 km. Ilmeisesti tämän vuoksi ruotsalaiset ovat ryhtyneet yhteistyöhön Kööpenhaminan yliopiston kanssa tarkoituksena tutkia veden virtaus- ja lämpötilaolosuhteita voimalaitoksen läheisyydessä.

Barsebäckin läheisyys useisiin kaupunkeihin nähden (tässä voidaan puhua jo lähisijoitusvoimalasta) on aiheuttanut sen, että on ryhdytty harkitsemaan mahdollisuutta käyttää jotain voimalayksikköä myös kaukolämmitykseen (lähinnä tulisi kyseeseen Lund). Päätöksiä ei ole vielä tehty, eikä kaukolämpökäyttö tule kyseeseen kahdessa ensimmäisessä voimalayksikössä.

## 6. RISÖN TUTKIMUSKESKUS

Ennen kiertokäyntiä Risön tutkimuskeskuksessa isännät esittelivät kuvin ja sanoin tutkimuskeskusta ja siellä tehtävää työtä. Kiertokäyntien jälkeen oli järjestetty cocktail- ja päivällistilaisuus, jossa oli mahdollisuus keskustella isäntien kanssa.

### 6.1 Vedenkäsittelylaitteisto

Ensimmäisenä kohteena esiteltiin vedentislauslaitteisto, joka tuotti erittäin puhtaaksi, steriiliksi ja pyrogeenivapaaksi tislattua vettä. Laitteen kiehutusosassa kehitetty höyry tiivistetään nostamalla painetta mekaanisella pumpulla, ja höyryn tiivistyessä syntyvä lämpö siirretään lämmönvaihtimella takaisin nesteeseen, joka prosessin seuraavassa vaiheessa joutuu höyrystymään. Saatu tisle johdetaan vielä toisiolämmönvaihtimeen, jossa se luovuttaa ylimääräisen energiansa syöttöveden esilämmitykseen. Tällä toimintaperiaatteella päästään pieneen energiankulutukseen eikä erillistä jäähdytysvesijärjestelmää tarvita. Samanlaista prosessia voidaan käyttää myös tehoreaktoreiden radioaktiivisten jätteiden käsittelyssä, jolloin jätteistä poistetaan vesi tällä menetelmällä ja syntyvät kiinteät ainekset varastoidaan.

## 6.2 Metallurgian laboratorio

Seuraavaksi kävimme rakennuksessa, johon oli sijoitettu "hot cell" -osastoja. Sitten tutustuimme metallurgian laboratorioon, jossa esiteltiin polttoainesauvojen suojakuoriputkien dimensio- ja defektimäärien tarkastuslaite. Tarkastusmenetelmä perustuu ultraäänen hyväksikäyttöön ja sitä sovellettiin zircalloy-putkiin. Laitteiston muodosti tarkastettavaa putkea ympäröivä vesikammio, johon oli sijoitettu ultraäänianturit. Koska vesikammio antureineen pyörii suurella nopeudella tarkastettavan putken ympäri, on tarkastusnopeutta saatu lisätyksi huomattavasti. Suurena etuna on se, että mekaanisilta kosketuksilta putken ja antureiden välillä on vältytty, joten putkelle ei tästä syystä aiheudu vaurioita. Putken liikkuessa antureiden editse mitataan sen dimensiot ja defektimäärä samanaikaisesti. Putken sisähalkaisija lasketaan mitatun seinämän paksuuden ja ulkohalkaisijan perusteella. Ultraääniantureista saatavat signaalit käsitellään tietokoneella, joka välittömästi laskee dimensiot ja defektien määrän. Dimensiot ja defektit on lajiteltu koon mukaan ja niitä verrataan spesifikaatioihin. Tietokone luokittelee sitten nämä tiedot putken eri osille.

Erityisesti laitteiston toimintanopeus ja tarkastuksen luotettavuus jäivät mieleen tästä mielenkiintoisesta esittelytilaisuudesta.

## 6.3 Kylmäneutronilaitteisto

Lopuksi esiteltiin rakenteilla oleva kylmäneutronilaitteisto. Kylmäneutronilähde eroaa sekä muodoltaan että jäähdytysperiaatteeltaan reaktorilaboratorion rakenteilla olevasta vastaavasta laitteesta. Pääasiallisena syynä lähteen huomattavan pieneen kokoon on DR 3-reaktorin aiheuttama voimakas gammalämmitys, mistä syystä moderaattorisäiliöön on käytetty mahdollisimman vähän materiaalia. Toinen mielenkiintoinen ratkaisu oli, että laitteistossa käytetään nestemäistä vetyä sekä jäähdytteenä että moderaattorina. Kylmäneutronispektrin optimoiminen tapahtuu säätämällä moderaattorisäiliön lämpötilaa, mikä moderaattorin lämpötilan lisäksi vaikuttaa sen tiheyteen. Laitteiston toimintavarmuutta lisää kahden kryogeneraattorin käyttäminen, minkä lisäksi turvallisuus varmistetaan sulkemalla koko systeemi heliumvaipan sisälle. Laitteiston on tarkoitus valmistua vuoden 1973 aikana.

## 7. ESITELMÄTILAISUUDET m/s FINLANDIALLA

### 7.1 Sandvik esittäytyy (1973-05-12)

Sandvikin edustajat esitelmä- ja keskustelutilaisuudessa olivat Sören Lönnberg, Jan-Christer Carlén, Hjalmar Depken, Kari Aro ja Åke Claesson.

Tervetuliaissanat lausui, ohjelman ja Sandvikin edustajat esitteli Sören Lönnberg.

Sandvikin valmistamien, ydinvoimalaitoksiin liittyvien komponenttien materiaaleista kertoi Jan-Christer Carlén. Lähinnä oli kyse reaktorin polttoainesauvojen suojakuorien, höyrystäjiin ja lämmönvaihtimiin liittyvien putkien ja reaktorin paineastiaan liittyvien komponenttien materiaalista. Eri materiaaleista jaettiin myös kirjallista tietoutta.

Seuraavana oli ohjelmassa n. 20 minuutin filmiesitys. Filmissä esitettiin putken valmistukseen liittyvät eri valmistus- ja tarkastusvaiheet.

Ydinvoimalaitosten putkimateriaalien laadunvalvonnasta ja ainetta rikkomattomista tarkastusmenetelmistä puhui Hjalmar Depken. Hän esitteli Sandvikin käyttämiä putken tarkastusmenetelmiä, jotka he ovat itse kehittäneet. Laadunvarmistuksessa painotettiin erityisesti organisaation tärkeyttä (tarkastus- ja valmistusosaston on oltava toisistaan riippumattomia.).

Tilaisuuden lopuksi oli ATS:n jäsenillä tilaisuus kysymyksien esittämiseen Sandvikin edustajille.

## 7.2 Esitelmätilaisuus 1973-05-13

Laivalla tapahtuvan paluumatkan viimeisen päivän sunnuntain 1973-05-13 esitelmätilaisuuden aloitti dipl.ins. Risto Tarjanne VTT:n lämmitysreaktoriryhmästä puhumalla aiheesta "Ydinenergia ja yleisö". Esitys pohjautui ko. aihetta käsittelevään Bo Lindellin kirjaan, Geneven neljännen kansainvälisen ydinenergian rauhanomaista käyttöä käsittelevän konferenssin (v. 1971) papereihin ja lehtikirjoituksiin. Mainittakoon, että juuri matkan aikana oli Ruotsin lehdissä julkaistu uutinen, jonka mukaan Ruotsin ydinvoimalaitossuunnitelmat voidaan jäädyttää siihen saakka kunnes laitosten turvallisuudesta on saatu varmat takeet. Ruotsin valtiopäivien asettama komitea oli nimittäin julkaissut mietinnön, jossa se ehdottaa hallituksen määräävän uusien ydinvoimalaitosten rakentamisen lopetettavaksi siihen saakka, kunnes radioaktiivisesta säteilystä aiheutuvat vaarat on täydellisesti tutkittu.

Dipl.ins. Tarjanne tarkasteli esitelmässään mm. erilaisia ydinenergian käyttöön liittyviä riskejä. Hän totesi mm. ydinenergiaa käytettäessä radioaktiivisen säteilyn lisääntyvän n. 10 % ja erilaisten onnettomuusriskien olemassaolon. Esitelmän jälkeen seurasi aiheesta erittäin vilkas keskustelu.

Aamupäivän toisessa esityksessä dipl.ins. Seppo Salmenhaara selosti VTT:n reaktorilaboratorion harjoittamaa tutkimus- ja koulutustoimintaa. Reaktorilaboratorio jakautuu käyttö-, säteilysuojelu-, neutronifysiikan, reaktorifysiikan ja isotooppitekniikan jaostoihin. Neutronifysiikan jaoston työstä esitelmöijä mainitsi mm. rakenteilla olevat neutronidiffraktori- ja kylmäneutronilaitteistot, joita voidaan käyttää raaka-ainetutkimuksissa. Reaktorifysiikan jaostossa kerrottiin suoritettavan reaktoriin liittyviä kohinatutkimuksia. Isotooppijaostossa mainittiin suoritettavan merkkiainetutkimuksia ja aktivointianalyysejä sekä valmistettavan lääketieteellisiä isotooppeja.

Iltapäivän ensimmäisessä esitelmässä dipl.ins. Tapani Graae Finnatomista esitteli valmistavan teollisuuden harjoittamaa tutkimustoimintaa. Koska teollisuuden tarkoituksena on valmistaa markkinoimiskelpoisia tuotteita, totesi esitelmöijä, ettei varsinaista perustutkimusta suoriteta, vaan laadunvarmistukseen liittyvät asiat ovat etualalla. Dipl.ins. Graae totesi kansainvälisestä yhteistoiminnasta olevan huomattavaa apua kotimaisen teollisuuden pienehköille tutkimusresursseille.

Dipl.ins. Graae kertoi, että valmistavassa teollisuudessa tutkimustoimintaa tapahtuu tuotekohtaisten kehitysprojektien yhteydessä, "perustutkimuksena", joka ei liity suoranaisesti tiettyyn tuotteeseen,ja myös erillistoimintana joissakin tapauksissa. Dipl.ins. Graae esitteli myös Finnatomin osakasyhtiöiden toimittamia ydinvoimalaitoskomponentteja kuvin ja sanoin.

Matkan viimeisenä esitelmänä LuK Simo Malkamäki Wallacista selosti ydinvoimalaitoksen säteilyvalvontajärjestelmää teollisena kehitysprojektina. Hän totesi, etteivät vanhat valmistuksessa olevat Wallacin säteilyvalvontalaitteet täyttäneet ydinvoimalaitoksessa niille asetettuja vaatimuksia. Niinpä Loviisan ydinvoimalaitosta varten jouduttiin käynnistämään kehitysprojekti, jossa koko säteilyvalvontajärjestelmä suunniteltiin alusta alkaen uudelleen.

Matkan päätteeksi suoritettiin esittelykierros, jossa jokainen osanottaja lyhyesti kertoi missä työskentelee ja mitä tekee.

## 8. YHTEENVETO

Kaikki matkalla olleet VTT:n tutkijat työskentelevät ydinvoimalaitoksiin liittyvissä projekteissa. Monella matkalla olleista tutkijoista ei kuitenkaan ole ollut aikaisemmin tilaisuutta tutustua rakenteilla olevaan tai toimivaan ydinvoimalaitokseen. Tällä matkalla sai ydinvoimalaitoksista varsin konkreettisen kuvan, sillä tutumiskohteina oli sekä valmis laitos että eri rakennusvaiheissa olevia laitoksia. Monet osanottajista eivät myöskään olleet aikaisemmin käyneet Ruotsin ja Tanskan ydintutkimuskeskuksissa. Niissä käynti oli erittäin mielenkiintoista, sillä niissä tehdään paljon samankaltaista työtä kuin matkan osanottajat itsekin tekevät.

Matkaan liittynyt alustettu esitelmä- ja keskustelutilaisuus paluumatkalla laivalla oli myös varsin mielenkiintoinen. Matka tarjosi myös mainion mahdollisuuden tutustua ydinvoimatekniikan alalla toimiviin sekä kotimaisiin että ulkomaisiin henkilöihin ja vaihtaa mielipiteitä heidän kanssaan.

SANDVIK
Jan-Christer Carlén

# ADVANCED MATERIALS FOR NUCLEAR POWER PLANTS

## INTRODUCTION

The power supply in the world will to an ever larger extent be based on nuclear power. In several countries nuclear power will account for the main part of the capacity increase during the 1970's. According to information published in May 1970 by The International Atomic Energy Agency, the installed nuclear capacity in 1971 amounts to about 35.000 MW, Fig. 1. The current plans show an increase until 1975 up to 123.000 MW and for 1980 up to about 300.000 MW.

For the manufacturers of reactor components and materials this expansion will be of great importance. At the same time this development will put very high demands on technical knowledge and delivery capacity.

Fig. 2 shows a principle sketch for a light-water reactor indicating some of the more interesting items from a metallurgist's point of view. These are

- fuel canning materials
- steel and nickel alloys for steam generators and other heat exchangers in nuclear power plants
- cladding materials for nuclear pressure vessels
- tubular products for pressure vessel components

## FUEL CANNING MATERIALS

In a nuclear reactor, the fuel has to be separated from direct contact with the coolant in order to prevent undesirable chemical reactions and spreading of radioactive decay products. The nuclear fuel, often consisting of sintered $UO_2$ pellets, is therefore contained in tubular fuel cans which are put together in bundles constituting the fuel elements. The canning material also serves to support the fuel and to define its geometrical shape which is of prime importance in carrying out a controlled nuclear fission process. A suitable canning material should

have a low absorption cross section for neutrons
provide adequate strength and ductility to contain and support the fuel during the period of burn-up
be compatible with the fuel, i.e. no harmful chemical reactions should occur at the canning-fuel interface
show good corrosion resistance to the coolant
show adequate heat conductivity

When neutron absorption, melting point and other essential design properties are taken into account, it is clear that only a few metals are suitable as canning materials.

## Factors affecting the choice of material

In order of increasing cross section for thermal neutrons the sequence of some interesting metals is as follows: Be, Mg, Zr, Al, Nb, Fe, Mo, Cr, Ni

For a thermal reactor based on natural uranium the metals beyond Al would have to be excluded because the neutron absorption cross section is too high. Only Zr and Al are reasonably corrosion resistant in water coolant but Al falls short because of its unsatisfactory mechanical properties. Zirconium (melting point 1855°C) is thus the predominant base metal for fuel cans for natural uranium in water-cooled thermal reactors and in thermal reactors with enriched fuel. In the latter instance, the higher neutron flux will permit the use also of thin-walled stainless steel cladding. However, the somewhat lower cost of stainless steel canning tubes is usually counter-balanced by the cost for loss of reactivity.

## Zirconium alloys

When zirconium was first tested for corrosion resistance in hot water and steam in the range 250 to 400°C, very erratic results were obtained. Further the mechanical properties of pure zirconium were unsatisfactory.

After long research and development work it was concluded that an acceptable combination of mechanical and corrosion properties could be obtained by adding small amounts of tin, iron and chromium. The corrosion resistance is further improved by adding some nickel, but this alloying element makes zirconium more prone to hydrogren pick-up. The compositions of zirconium alloys used in today's canning are given in Table 1. The alloys are designated Zircaloy 2 and Zircaloy 4.

During operation, the canning tube shall have sufficient strength to resist external overpressure from the coolant. Later on it must resist internal overpressure from gaseous fission products. Also as the burn-up increases, there is a certain swelling of the $UO_2$-fuel. Both these expansions create a hoop stress which is applied very slowly. The deformation necessary to meet these expansions should be considered as a creep deformation under fluctuating load.

$UO_2$ is a fairly poor conductor of heat and very steep thermal gradients arise in the fuel. There is also a temperature gradient through the canning and its protective oxide film. The maximum surface temperature for Zircaloy cladding is at present about 320°C for BWR:s and somewhat higher for PWR:s. Occasionally, however, higher temperatures may occur (up to 400°C).

In a reactor fuelled with natural uranium the coolant can be heavy water, $D_2O$. With enriched fuel, light water is used. The coolant can be pressurized or boiling. In a pressurized water reactor (PWR) the hydrogen pick-up is higher than in a boiling water reactor (BWR). In pressurized systems the preferred cladding material is Zircaloy 4 which has an inherently lower hydrogen pick-up than Zircaloy 2.

To make tubes with the required combination of ductility and high-temperature strength is one of the most difficult problems to solve in the production of canning materials of zirconium alloys.

The canning tubes are cold worked to final dimension. The way in which the final dimension is obtained as well as the way of heat treatment are decisive for the mechanical properties of the tubes. By proper choice of type of rolling mill, dimensions of mandrels and rolls, reduction per cold pass, annealing time and temperature, etc., such a texture in the tube wall structure is obtained as to give the specified combination of yield strength and ductility.

Hydrogen in zirconium

Zirconium metal has a high affinity to hydrogen. In water-cooled reactors Zircaloy canning tubes take up hydrogen from the water during the corrosion reaction.

$$Zr + 2H_2O \longrightarrow ZrO_2 + 4H$$

In pressurized water reactors (PWR) the hydrogen pick-up is higher than in a boiling water reactor (BWR). Because of this, in PWRs the preferred cladding material is Zircaloy 4.

During cooling of the reactor from service temperature plate-like zirconium hydride particles are precipitated. These are most often aligned parallelly. A tangential orientation is obtained by using a high reduction of wall thickness with a minor change in tube diameter. A radial orientation on the other hand results from a large reduction of tube diameter.

The hydride plates may act as cracks, when the tubing is subjected to mechanical load. Under out-of-pile conditions such cracks will probably not propagate, but when the Zircaloy has been hardened and embrittled by neutron irradiation the cracks propagate easily. In this situation it is specially deleterious to have radially oriented hydrides, inducing the cracks to go straight through the wall thickness.

This phenomenon is the basis for the customer specifications for tangentially aligned hydrides. The specification is met by the canning-tube producer by balancing the reduction of wall thickness to the reduction of diameter during the tube rolling process.

Future zirconium alloys

The higher the temperature of the coolant the better the thermodynamic efficiency for power generation. For canning materials on a zirconium basis it is not possible to increase the service temperature appreciably. However, an increase of 20 or 40°C in comparison with the present Zircaloys may still be valuable. This might be obtained by adding more or new alloying elements to zirconium in order to raise the high temperature corrosion resistance and the high temperature strength.

A number of alloys have gone through the sequence of being developed, being evaluated out-of-pile, being the focus of general attention for some time, and being transferred to extensive and time consuming in-pile testing.

There is the Zr-2,5Nb-alloy which was first developed in Russia. It has higher strength and less tendency to hydrogen pick-up than Zircaloy-2. However, its corrosion resistance is not as good as for Zircaloy-2. So far it has been used mainly for pressure tubes in Canadian Candu reactors.

A further development of this alloy is the German Zr-3Nb-1SN which has high strength and good ductility and compares to Zircaloy 2 as regards corrosion resistance. However, serious disadvantages are a complicated manufacturing process and a high absorption cross section, which makes the use of enriched fuel necessary.

Another Russian zirconium alloy is Ozhenite-0.5 which also has attracted interest especially in Canada. Its composition is Zr-0.2Sn-0.1Fe-0.1Ni-0.1Nb. It is a soft and ductile alloy which displays higher corrosion resistance and lower hydrogen pick-up than Zircaloy-2.

Zr-1.15Cr-0.07Fe is an alloy developed in the USA in order to reach the best high temperature corrosion resistance in BWRs while its mechanical properties are similar to Zircaloy 2 and 4. However, in-pile corrosion tests have given varying results for this, from the start, very promising alloy.

Another alloy which only recently has been considered in the West is the Russian Zr-1Nb, used as canning in the majority of Sovjet BWRs. It is somewhat softer and more ductile than Zircaloy-2 and is said to have good corrosion resistance.

## Stainless steels and nickel alloys for canning tubes

Austenitic stainless steels of conventional types have been used in various water-cooled thermal reactors with canning temperatures around 300°C. There is, of course, a strong incentive to increase the power density of the fuel rod and also to increase the temperature of the coolant. This will result in higher canning temperatures and above about 350°C the stainless steels and nickel alloys should therefore be chosen.

## Canning materials for fast breeder reactors

For the fast breeder reactors which are under development and in consideration as the second generation of nuclear power plants as the most economical type around 1980, the material problems for the core components are even more pronounced than for the thermal reactors. For canning tubes special austenitic stainless steels and vanadium alloys are being in consideration and under development and testing. Although this is a technically interesting and difficult area of development, it would lead too far within the scope of this article to deal in detail with this subject.

# HEAT EXCHANGER TUBES IN NUCLEAR REACTORS

Material supplied by the Sandvik Steel Works to nuclear power stations includes heat exchanger tubes of stainless steels and nickel alloys. In order to meet the requirements on these tubes, a special tube mill has recently been built. Fig. 3 shows an interior view from this mill.

Depending on the type of reactor, which means differences in reactor coolant and steam temperature, the choice of tube material in the heat exchangers will vary. In the following the presentation of suitable tube materials for heat exchanger applications will be referred to different reactor types.

Steam generators (Table 2)

Water-cooled reactors

In pressurized water reactors the steam generator tubing operates in water and steam in the range 250-325°C. Since corrosion products in the primary circuit are to be avoided, stainless austenitic steels or nickel alloys should be used. Apart from general corrosion resistance, the resistance to stress corrosion, pitting, and intergranular corrosion has to be considered.

With the risk of stress corrosion attacks due to possible chloride and oxygen contaminations in the coolant circuits, however, alloys with high nickel contents have been preferred to conventional austenitic steels. Two types of stress corrosion may occur under pressurized water conditions resulting in transgranular and intergranular cracks respectively. The transgranular attack may occur in austenitic steels in the presence of chlorides even at very low chloride contents and it is generally accepted that nickel increases the resistance to this type of corrosion. Therefore, Alloy 600 with more than 72 % Ni is being used to a large extent as steam generator tube material.

As regards the resistance to intergranular stress corrosion, the suitability of Alloy 600 has been doubted in recent years. In fact, French and British research work has shown that even in demineralized water intergranular attacks can occur in nickel-base alloys like Alloy 600 after about 10.000 hrs at 350°C. Such a behaviour must be considered as serious, as it cannot be avoided even by a proper control of water chemistry. A number of steam generator tube failures have also occurred in practice, both with Alloy 600 and 18/8 steel tubes.

The experience with Alloy 600 referred to has led to Alloy 800 being preferred in several cases. The intermediate nickel content of 32.5-35 % in this alloy seems to offer a satisfactory resistance to both transgranular and intergranular attacks. In the Sovjet Union 18/8-steels are being used.

Gas-cooled reactors

In gas-cooled reactors the steam generator tubing is exposed to carbon dioxide or helium as the heat delivering medium and to water or steam as the heat absorbing medium. The tube wall temperature may reach 650°C. The choice of material may in principle be based on the same considerations as for a conventional boiler but due consideration must be given to the corrosiveness of the coolant gas (He or $CO_2$).

In cases where reliable control of the chloride or oxygen content of the feed water cannot be guaranteed, it may be necessary to select steels with good resistance to stress corrosion also for the superheating section, especially in connection with shutdowns when wet steam might be present. SANDVIK Sanicro 31 (Alloy 800) is a material, which may provide this. This grade has been selected for HTR:s in the United States and Germany. Of course, also the ferritic steels are insensitive to this type of corrosion.

Sodium-cooled reactors

In fast reactors using sodium as a collant, the heat exchanger tubing is exposed to sodium at 600-600°C on the primary side and to water and steam on the secondary side. Both ferritic and austenitic steels may be used. Conventional low-alloyed ferritic steels present a problem because sodium tends to decarburize them with carburization of austenitic steels in the cirquit as a consequence. This has led to the development of Nb-stabilized ferritic steels, e.g. the alloy containing 2.25 %

As the installations are working at high pressures it may be necessary to resort to austenitic steels or nickel alloys which possess higher hot strength than ferritic steels. The risk of stress corrosion must also be considered in some cases and here an austenitic alloy with low susceptibility to stress corrosion such as Alloy 800 may be selected. The variant of Alloy 800 designated Sanicro 30, with its good resistance to sensitizing and intergranular corrosion, is, as a matter of fact, being considered for use in the entire steam generating system.

<u>Other heat exchangers</u> (Table 3)

Feed-water heaters are used in all large power stations being built today. In the nuclear field the austenitic steels, especially of the type AISI 304 (SANDVIK 5R10) or 304L (SANDVIK 3R12), are preferred in direct-cycle power reactors because of their high general corrosion resistance. Also for applications in other types of nuclear reactors, there is growing interest in stainless austenitic steels.

Sandvik has facilities to meet all the requirements on heat exchanger tubes. Tubes in lengths up to about 35 m (115 ft) can be produced in all common as well as many special stainless steels and alloys. Such long tubes are normally delivered as U-bends or in other configurations.

<u>Power plant condensers</u>

During the last few years, development in the condenser field has shown a tendency towards larger units and increasing demands for reliability in service. More and more stainless steels condenser tubes have come into use instead of tubes of other alloys. For condensers we may thus expect a wider use of stainless steel tubes and perhaps also of titanium tubes. In ascending order of corrosion resistance <u>Table 4</u> specifies a few types of materials which may be considered. For safety reasons a Mo-alloyed steel should normally be chosen in order to avoid corrosion during shut-down periods. With increasing temperature and increasing contents of chlorides and micro-organisms in the cooling water, steels with high Mo-contents should be preferred, for instance grade SANDVIK 2RN65 which contains 4.7 % Mo or, still better grade SANDVIK 2RK65 with 4.5 % Mo and 1.5 % Cu showing further improved resistance to crevice corrosion. Titanium would then be a further step towards improved reliability in service.

CLADDING MATERIALS FOR PRESSURE VESSELS

For certain parts of the reactor, the requirements on the construction material are so diversified that they could not, practically or economically, be met by one single grade. For the pressure vessel, for instance, a low-alloyed pressure vessel steel offers great advantages by its good tensile properties and the relatively low price. On the other hand, this type of steel does not possess the required corrosion resistance. In this respect, an austenitic stainless steel performs adequately. However, a pressure vessel entirely of stainless steel would become expensive and necessitate a heavy wall thickness owing to the relatively low mechanical strength of this material.

The problem is mastered by using low-alloyed steel for the heavily stressed shell of the pressure vessel and by lining it with a stainless steel withstandning the corrosive attack from the moderator or the coolant.

The commonest method for applying the corrosion protective lining of a nuclear pressure vessel is cladding with stainless weld metal. In principle, the cladding operation can be executed by any of the common arc welding processes. When large surfaces are involved, the submerged arc welding process is usually preferred, mainly owing to the high rate of deposition as compared to other welding processes.

A variant of the submerged arc cladding process employs a consumable strip electrode, Fig. 4.

Sandvik produces a series of grades for various applications of the cladding methods, see Tables 5 and 6. SANDVIK 3RE13 is an overalloyed grade of the AISI 309-type, with 24 % Cr and 13 % Ni, especially suitable for applying the buffer layer direct on the base metal. SANDVIK 2R16 and 6R42 are intended for the top layers in the multi-layer technique and their analyses are adjusted to meet the specifications for 304L and 347 respectively. The other stainless steels are primarily intended for single layer cladding resulting in analyses corresponding to AISI 304L or 347.

Another interesting nuclear application of the cladding process is the overlaying of Ni-Cr-Fe alloy on the tube plates of steam generators which have to withstand large cyclic temperature and pressure varianions. The cladding of the tube sheets as well as the welding of the tubes to the clad layer are extremely exacting operations which put severe requirements on the cladding material.

TUBULAR PRESSURE VESSEL COMPONENTS

The design of the pressure vessel for water-cooled reactors involves the choice of material for the control rod inlet tubes and other studs connected to the reactor vessel. As the corrosion resistance requirements on the inner surface of these tubes are the same as for the pressure vessel cladding, materials with at least the same corrosion resistance as AISI 304 have to be used.

The use of austenitic steel for control rod inlet tubes has not been quite successful, however, and failures have occurred. One problem is the joining of the austenitic tube to the ferritic vessel head. Furthermore, the great difference in thermal expansion between these components causes high local stresses. These disadvantges could be avoided to some degree by the use of Alloy 600 (SANDVIK Sanicro 71). However, further advantages are gained by using a composite tube consisting of a ferritic tube with a lining of austenitic steel. By co-extrusion of the two material components a perfect metallurgical bond can be obtained.

Fig. 5 shows how a composite tube is applied as a control rod inlet. If required, a fillet weld can be made between two ferritic components which greatly reduces the thermal stresses in the weld zone both during the welding procedure and in service. Another important advantage of the composite tube is that for the ferritic part of the tue a steel can be chosen which has a higher strength than the conventional austenitic steel especially as the control rod inlets often are subjected to fatigue stresses.

On other tubular pressure vessel components exacting requirements are often put as regards surface finish after machining good fatigue properties and general cleanliness, necessitating the use of high-vacuum remelted or electro-slag refined materials.

## CONCLUSION

Extensive research and development work is being carried out all over the world in order to improve existing nuclear power plant design and to develop new types of reactors with improved operation economy. Proper material selection including the development of especially designed materials for nuclear applications is a must for technical advance in this field. This calls for still closer cooperation in the future between all bodies concerned with reactor design and operation and their material suppliers.

| Zircaloy | Sn | Fe | Cr | Ni |
|---|---|---|---|---|
| 2 | 1.50 | 0.12 | 0.10 | 0.05 |
| 4 | 1.50 | 0.21 | 0.10 | ≤0.007 |

Table 1

Composition in weight per cent of zirconium alloys

STEELS AND NICKEL ALLOYS FOR STEAM GENERATORS IN NUCLEAR POWER PLANTS. COMPOSITION IN WEIGHT PER CENT

| SANDVIK | Corresponds to | C | Cr | Ni | Other elements |
|---|---|---|---|---|---|
| 3R12 | 304L | ≤0.030 | 18.5 | 10.5 | |
| 8R30 | 321 | ≤0.08 | 17.5 | 10.5 | Ti |
| Sanicro 30 | Alloy 800 | ≤0.030 | 21 | 34 | Al, Ti |
| Sanicro 70 | Alloy 600 | ≤0.030 | 16 | ≥72 | Al, Ti, Fe |
| HT7 | Grade T9 | 0.1 | 9 | - | Mo |
| HT9 | - | 0.2 | 11.5 | 0.5 | Mo, W, V |
| 7R60 | 316H | ≤0.07 | 16.5 | 13.5 | Mo |
| Sanicro 31 | Alloy 800 | 0.06 | 21 | 31 | Al, Ti |
| HT8X6 | - | 0.08 | 2.3 | 0.6 | Mo, Nb |

Table 2

STEELS AND NICKEL ALLOYS FOR FEED WATER HEATERS AND REHEATERS IN BWR SYSTEMS. COMPOSITION IN WEIGHT PER CENT

| SANDVIK | Corresponds to | C | Cr | Ni | Other elements |
|---|---|---|---|---|---|
| 1C342 | 430+Ti | 0.08 | 17 | - | Ti |
| 3R12 | 304L | ≤0.030 | 18.5 | 10.5 | |
| 5R10 | 304 | ≤0.05 | 18.5 | 9.5 | |
| 8R30 | 321 | ≤0.08 | 17.5 | 10.5 | Ti |
| 6R40 | 347 | ≤0.06 | 17.5 | 11.5 | Nb |
| Sanicro 30 | Alloy 800 | ≤0.030 | 21 | 34 | Al, Ti |

Table 3

| SANDVIK | Corresponds to | C | Cr | Ni | Mo | Cu |
|---|---|---|---|---|---|---|
| 5R10 | 304 | ≤0.05 | 18.5 | 9.5 | - | - |
| 5R60 | 316 | ≤0.05 | 17.5 | 13.5 | 2.7 | - |
| 3R64 | 317L | ≤0.030 | 18.5 | 14.5 | 3.5 | - |
| 2RN65 | - | ≤0.020 | 17.5 | 24 | 4.7 | - |
| 2RK65 | - | ≤0.020 | 19.5 | 25 | 4.5 | 1.5 |

Table 4
Steels for condenser tubes. Composition in weight per cent.

CLADDING MATERIALS FOR PRESSURE VESSELS
COMPOSITION IN WEIGHT PER CENT

| SANDVIK | C | Cr | Ni | Nb | Others |
|---|---|---|---|---|---|
| Strip | | | | | |
| 3RE41 | ≤0.025 | 24 | 12.5 | 0.8 | |
| 3RE14 | ≤0.025 | 24 | 12.5 | - | |
| 3R42 | ≤0.025 | 20 | 10 | 0.6 | |
| 2R16 | ≤0.020 | 19.5 | 10.5 | - | |
| Sanicro 72 | ≤0.030 | 20 | 72.5 | 2.5 | Mn 3.0<br>Fe ≤1.0 |
| Wire | | | | | |
| 2X1RE13 | ≤0.020 | 25 | 12 | - | |
| 3RE13 | ≤0.025 | 24.5 | 12.5 | - | |
| 3RE41 | ≤0.025 | 24 | 12.5 | 0.8 | |
| 3X4R17 | ≤0.030 | 22 | 11 | - | |
| 6R42 | ≤0.05 | 19.5 | 9.5 | 10.6 | |
| 2R17 | ≤0.020 | 20 | 10 | - | |
| Sanicro 72 | ≤0.030 | 20 | 72.5 | 2.5 | Mn 3.0<br>Fe ≤1.0 |

All grades with Co ≤ 0.05%

Table 5

Table 6

CLADDING MATERIALS AND SUGGESTED TYPES OF FLUXES

| Required weld metal ASTM type | Form of filler | Two layers First | Second | One layer | | |
|---|---|---|---|---|---|---|
| 304L | Strip | 3RE14 Cr-comp or neutral | 2R16 Cr-comp | 3RE14 neutral or Cr-comp | 3RE15 Cr-comp or neutral | 2R16 alloying |
| 304L | Wire | 3RE13 Cr-comp or neutral | 2R17 Cr-comp | 2X1RE13 neutral | 3RE13 Cr-comp or neutral | 3X4R17 Cr-comp |
| 347 | Strip | 3RE14 Cr-comp or neutral | 3R42 Cr-comp | 3RE41 neutral or Cr-comp | 3RE47 Cr-comp or neutral | 3R42 alloying |
| 347 | Wire | 3RE13 Cr-comp or neutral | 6R42 Cr-comp | 3RE41 Cr-comp | | |
| 316L | Strip | 3RE14 Cr-comp or neutral | 2R63 Cr-comp | 2R63 alloying | | |
| 316L | Wire | 3RE13 Cr-comp or neutral | 2R62 Cr-comp | | | |
| 420 | Strip Wire | 6C27 Cr-comp | 6C27 Cr-comp | | | |
| 16Cr/72Ni Alloy 600 | Strip Wire | Sanicro 72 Low Si | Sanicro 72 Low Si | | | |

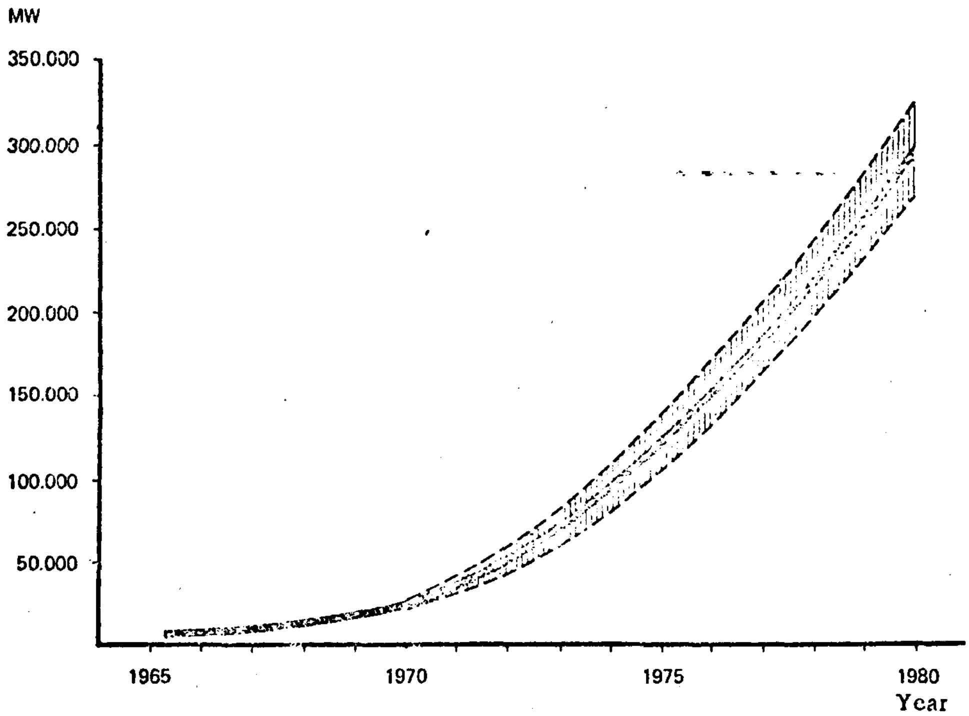
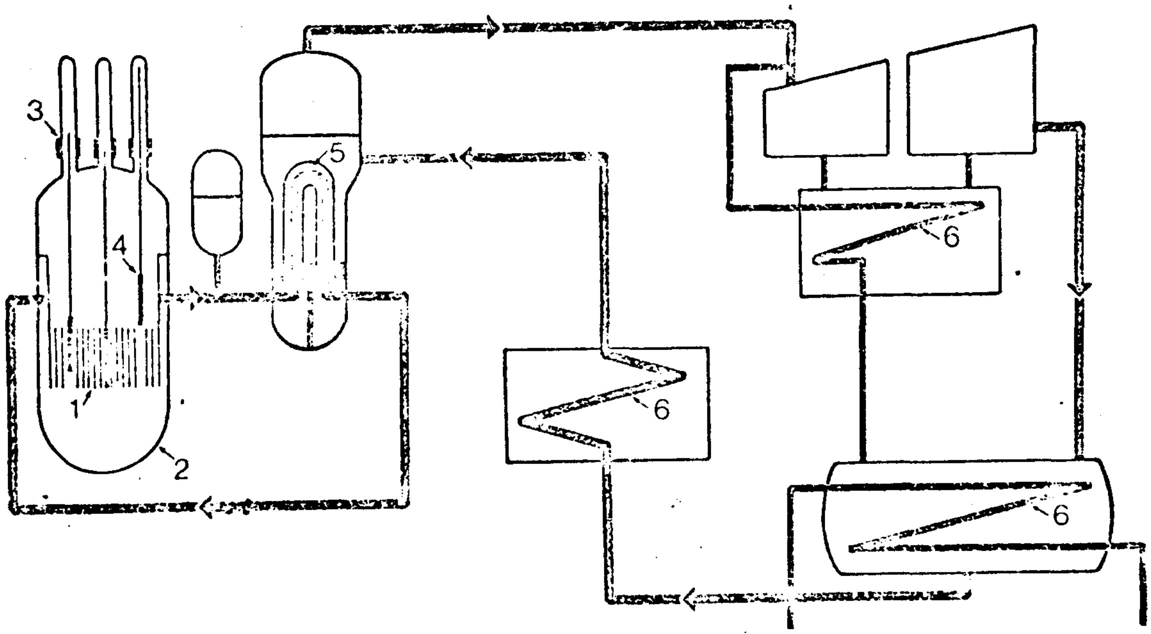



Fig. 1
Development of nuclear power up to 1980
Mean value of various estimations




1 Canning tubes Zircaloy
2 Overlay welding stainless steel
3 Control rod inlets, composite tubes
4 Control rods Zircaloy and stainless steel
5 Steam generator tubes Sanicro
6 Stainless steel tubes

Fig. 2
Pressurized water reactor, PWR

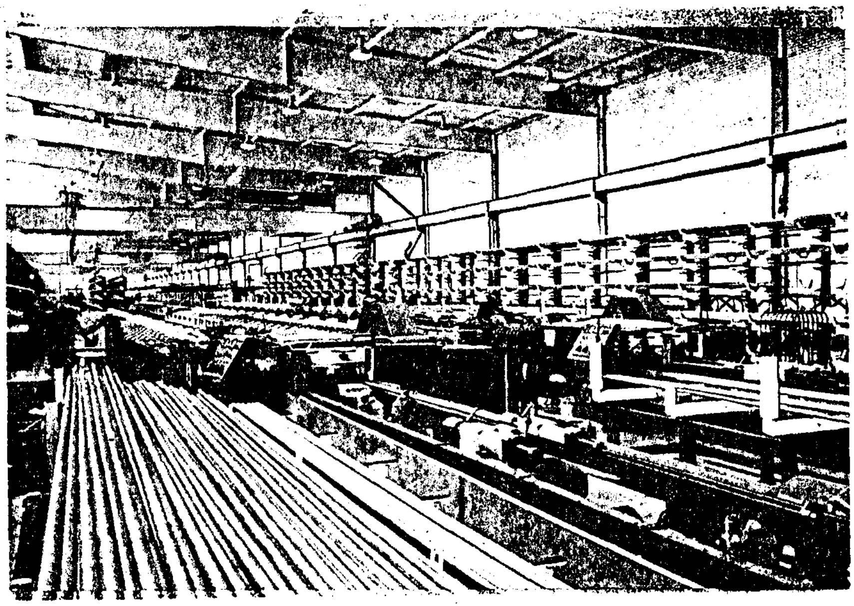


Fig. 3
View of mill for heat exchanger tubing

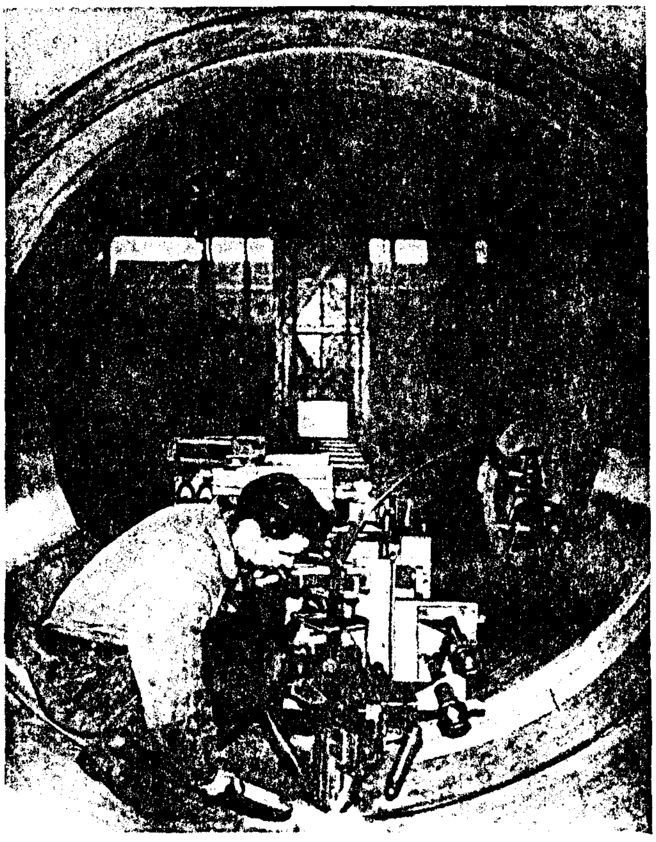


Fig. 4
Strip cladding equipment

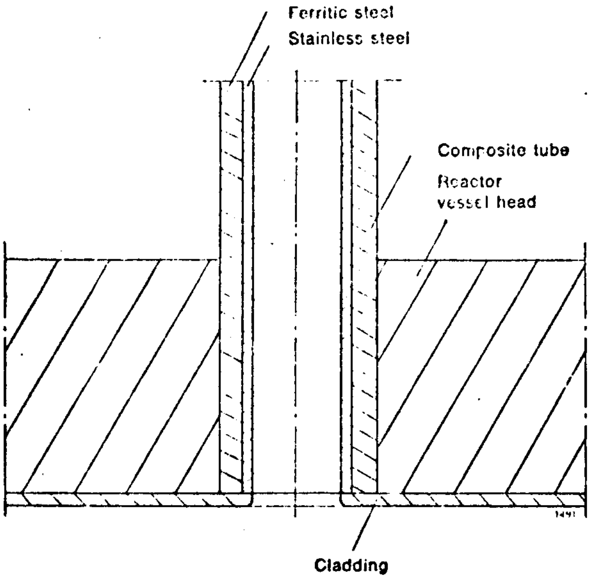



Fig. 5
Composite tubes for control rod inlets

| SANDVIK Steel Research Center | Progress Report | T No. 62255<br>Date 1972-02-07<br>Projekt No. F41<br>Author TR/Carlén/IJ |
|---|---|---|

## Non-destructive Testing of High Quality Tube and Pipe for Advanced Applications

### 1. Introduction

The demands put on tubular products today for nuclear and other critical applications have been made possible to meet by a simultaneous development of production and testing methods. If today's facilities and requirements for non-destructive testing had been available a couple of decades ago, the result would certainly have been catastrophic as regards yields and economy for a tube-maker. It has thus been necessary to refine the production and testing methods in parallel to fulfill the increasing demands for reliability and proper function of tubular components in nuclear, chemical and other industries. The aim of this paper is to give a tube-maker's view on advantages and disadvantages of various methods and procedures, rather than to cover the whole field of non-destructive testing. To keep the paper within reasonable limits, tha analysis will mainly concern heat-exchanger tubing in the size range 1/2"- 1 1/2" outer diameter.

### 2. Ultrasonic Examination - Shear Waves

The most common way of ultrasonic examination of tubes is immersion testing using shear waves. As coupling medium between the transducers and the tubes water, often demineralized, is normally

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used. Furthermore it can be stated that the testing is usually made against a calibration tube with artificial notches as standard or reference defects. In most other respects the testing conditions may vary depending on type of equipment and the specification and testing procedure applied. To discuss basic acoustics and electronics would lead too far and the intention is rather to draw the attention to some fundamental factors determining the testing sensitivity and procedure.

## 2.1. Calibration standard and testing sensitivity

Modern equipment for ultrasonic examination of heat exchanger tubing utilizes line-focused transducers (common transmitters-receivers). The ultrasonic beam should ideally have a constant sound intensity over the entire focus length and for approximate calculations of the testing sensitivity, such an assumption can be made. The calibration tube shall according to most specifications contain V- or ⊔-shaped artificial notches, the shape, size and tolerances of which are generally well defined. To arrive at a constant testing sensitivity one of the following two conditions must be fulfilled:

$$L \geq l + s \quad \text{and } l \geq 2s \qquad (1)$$

or

$$l \geq L + s \quad \text{and } L \geq s \qquad (2)$$

where

$L$ = length of artificial notch

$l$ = focus length

$s$ = pitch

When testing according to (1) the sensitivity is determined by the focus length and the depth of the artificial notch. The tube surface is examined with minimum 50 % overlap from revolution to revolution, which affords good possibilities to detect defects shorter than the focus length.

When testing according to (2) the sensitivity is determined by the dimensions of the artificial notch. Also this method gives minimum 50 % overlap to enable the detection of defects as short as or shorter than the artificial notch.

It can thus be stated that the artificial notch dimensions in the calibration tube is only one factor affecting the testing sensitivity. To get a complete picture of the testing sensitivity the whole testing procedure must be well defined.

## 2.2. Influence of defect shape and orientation

Most specifications for ultrasonic examination prescribe testing for longitudinal defects only, i.e. with the ultrasonic beam traveling around the tube and with only longitudinal artificial defects in the calibration tube. For large diameter, heavy wall tubing, where the relation between the artificial notch depth and the tube wall is low, this may be acceptable, but for thin wall heat exchanger tubing, this testing method is often not sensitive enough. Defects of a transverse nature may cause low or negligible defect signals although they are deep and extended. But also defects with extension in the longitudinal direction may pass undetected because their shape is such that the ultrasonic echo will be weak and scattered. For such tube dimensions, when the application requires highest possible quality, the ultrasonic examination should be performed for both longitudinal and transverse defects to eliminate the influence of defect orientation on the test result. Furthermore, the examination should be performed by scanning in two directions (clockwise and counter-clockwise for longitudinal defects, with and against the tube for transverse defects respectively) to eliminate the influence of defect shape and orientation as far as possible.

The ideal non-destructive testing method should be quantitative, i.e. give information of the size of a possible defect. Ultrasonic examination can in general be said to be semi-quantitative, as the defect signal magnitude depends on the size and shape of the defect cross section. By the above-mentioned procedure, ultrasonic examination has been made as quantitative as possible. However, also the location of the defect has some influence and the testing equipment should be calibrated to give as equal signals for internal and external defects as possible and using a calibration tube with both external and internal notches.

## 2.3. Acceptance criteria

In most cases all tubes, which do not produce a defect signal greater than the signal from the artificial notch or notches, are deemed to have passed the test. In discussing small size tubing where the possible defect size may form a significant portion of the wall thickness, the accuracy of the whole system may be of some importance for the interpretation and evaluation of the test results. To eliminate the influence of this uncertainty, the acceptance level is often defined as a certain percentage of the signal magnitude from the artificial notches. E.g. for PWR steam generator tubing, the acceptance level can be set to 70-90 % of the lowest signal from the artificial notches, depending on the whole testing procedure.

...5

## 2.4. Influence of structure on the test results

Although ultrasonic examination is a selective method and a reflecting defect area is necessary to produce a defect signal, the structure may influence the results. Thus, when testing to very high sensitivity, signals from grain boundaries may disturb the test results and hide possible signals from real defects due to "grassing" or background noise on the records. The likelihood for grassing to occur depends on testing frequency and sensitivity as well as the grain size. For austenitic steel and nickel alloy tubing the critical average grain size may vary from ASTM 3 to ASTM 7 depending on testing procedure. Primarily an irregular grain size with both coarse and fine grains may cause problems.

## 2.5. Some practical aspects

Ultrasonic examination is a slow and, therefore, expensive testing method when the requirements on testing sensitivity are high, especially when testing small size tubing with a low weight per unit length. The pitch is determined by the testing sensitivity and when the tubes have to be rotated, the rotation speed must be kept rather low to avoid vibrations and damage on the tubes. Since some years equipment with rotating transducers is available on the market, making far higher testing speeds possible. When the number of revolutions per minute can be raised from about 1.000 to 3.000 this has, of course, a significant influence on the testing costs.

To keep testing costs down, the equipment should be fully automatic. This means automatic colour marking of defective areas and automatic segregation of rejected tubes for repair or scrapping. When requested, the testing equipment shall, besides the electronic monitoring system, provide facilities for analogue recording of the test results. In future we can expect on-line computer treatment of all the information from ultrasonic examination which will facilitate interpretation, evaluation and statistical follow-up of the test results.

When looking at costs for ultrasonic examination it is also essential to bear in mind that the rejects caused by testing to a high sensitivity often are more expensive than the testing itself.

## 2.6. Example of ultrasonic testing specification

To illustrate the above aspects on ultrasonic examination it may be interesting to review a condensed specification, which is typical for PWR steam generator tubing at The Sandvik Steel Works.

Tube size and grade: 7/8"x0.050", Sanicro 30 or 70 (Alloy 800 or 600)

Equipment and method: Rotating barium titanate transmitter/receiver transducers, 4MHz, line-focused. Two transducers each for longitudinal and transverse defects, scanning in opposite directions.

Calibration standard: Artificial longitudinal and transverse defects (V-notch), internally and externally. Depth = 0.004" and length = 1/2". Automatic triggering for defect signals bigger than 75 % of the artificial notch indications. Recalibration after every 20 tubes.

Testing conditions: Focus length = 0.32", pitch = 0.16", longitudinal speed = 40'/min (3000 rpm)

Acceptance criteria: All tubes not containing defects that will trigger the monitor are deemed to have passed the test.

Fig. 1. shows the ultrasonic equipment used for this type of testing.

Testing against this specification has proved to be very satisfactory, which has been demonstrated by numerous metallographic examinations of tube sections showing defects which have been either rejected or accepted in the ultrasonic test.

## 3. Ultrasonic Examination - Longitudinal Waves

Ultrasonic examination with longitudinal sound waves is used primarily to measure the wall thickness of tube and pipe. So far this testing technique has mainly been used for manual checking of the wall thickness after local grinding of defects. However, for several years continuous wall-thickness measuring and recording has been applied for canning tubes for nuclear fuel and will certainly also be requested for heat exchanger tubing for service at high pressures in the future, especially when a tube failure may cause health hazards by spreading radioactive or toxic products. The principle for wall thickness measuring is shown in fig. 2.

Longitudinal waves are also used for checking the quality of a new type of tubular product getting more and more attention for applications in nuclear and chemical industry: co-extruded composite tubing. Such tubes are used, where the material requirements are conflicting and cannot be satisfied by one material only, e.g. as regards corrosion and strength or general corrosion and stress corrosion resistance simultaneously. The co-

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extrusion process gives composite tubes with a metallurgical bond between the two components and ultrasonic examination is used to check that such a bond is present all along the tube length. In this case a calibration standard is used, usually in the form of a flat-bottomed drill hole with a defined bottom area, situated at the bond. Also for this application ultrasonic examination has proved to be a very sensitive and accurate testing method.

4. Eddy Current Examination

The most common eddy current testing method for heat exchanger tubing is to utilize surrounding coils. With this testing technique inhomogeneities in the material in the form of defects and variations in dimensions and physical properties such as structure and stress level will be indicated. The main difficulty with this testing method is to distinguish indications of real defects from indications of other origin which are of no importance for judging the tube quality. Many attempts have been and are being made to try to separate those indications coming from defects only but so far no acceptable system is available for production testing. However, eddy current testing has the advantage of being a very fast and inexpensive testing method and is suitable for detection of such short defects, which may be difficult to find by ultrasonic examination, especially when testing only for longitudinal defects.

For welded heat exchanger tubing in stainless steels and nickel alloys eddy current examination is the most suitable non-destructive testing method as ultrasonic examination is impossible due to the structure and the geometrical shape of the weld. Welded and drawn tubes may, however, also be ultrasonically tested.

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## 4.1. Calibration and sensitivity

Most calibration standards for eddy current examination are based on drilled holes, either straight through the tube wall or to a certain depth from the outside, to set the sensitivity. As the testing sensitivity will always be higher at the outside of a tube and the drilled holes cannot be used for checking the sensitivity for internal defects, it is also common to use circumferential grooves on the internal and external tube surfaces. These are then used to select frequency and phase angle for optimizing the relation between internal and external defect sensitivity.

## 4.2. Practical aspects

As already indicated eddy current testing is sensitive not only to defects but also to volume changes in the cross section caused by minor wall thickness variations. This means that the manufacturing method chosen for making the tubes has a large influence on the applicable testing sensitivity. Cold-rolling (cold-pilgering) of tubes is an excellent method for tube production enabling high deformation rates and minimizing of excentricity but the discontinuous feed causes minor wall thickness variations (waviness), in the order of 0.0005"-0.0015". This waviness, which falls well within the specified tolerances may cause serious grassing or background noise. Tube drawing is more favourable in this respect but does not produce a better tube quality, rather the opposite as regards concentricity.

In testing of welded tubing in austenitic stainless steel, which usually shows excellent wall thickness tolerances, the background noise level may be rather high, even when the weld bead is very smooth. This may be caused, for example, by variations in ferrite content in the weld and stress levels (cold work) from the straightening operation.

## 5. Liquid Penetrant Examination

For tubing the liquid penetrant test is a somewhat questionable non-destructive method as mostly only the external surface can be checked. In this context it would lead too far to discuss all the variants of penetrant testing and I shall rather concentrate on the advantages and disadvantages of this technique in general.

The main advantage with dye penetrant examination of tubing is that injurious surface defects are easily detected also in sizes where other non-destructive testing methods are too insensitive. This refers mainly to large-diameter, heavy-wall tube and pipe, where even ultrasonic testing to a 3 % defect level may leave superficial laps, seams, etc. undetected. In such cases it may also be possible to examine the bore. Such high demands on the tube surface are often put on large diameter pipe for control rod operating systems, e.g. in sizes around 4-8" outer diameter and 3/4"-1 1/2" wall thickness. The reason is to eliminate defects which may serve as stress raisers or initiate corrosion attack. For heat exchanger tubing, however, these aspects are rather irrelevant.

Moreover, dye penetrant examination may be applied on manipulated tubing, e.g. U-bends or tubes bent to other configurations where neither ultrasonic, nor eddy current examination can be applied.

The main disadvantage of liquid penetrant testing is the rather inadequate information. From possible indications it is very difficult to tell anything about the depth and harmfulness of the defects. Therefore, the elimination of harmless, very superficial defects may lead to unnecessary and costly repair work.

As a serious drawback must also be mentioned that liquid penetrant testing is a slow, dirty and costly operation requiring large space, especially for handling long length tubing. For heat exchanger tubing, already tested to a strict ultrasonic specification, the method does not give a significant increase in quality in relation to the costs involved, and should in most cases not be applied.

At The Sandvik Steel Works dye penetrant examination is primarily used for heavy wall tubing where ultrasonic or eddy current examination is not enough sensitive for superficial defects and for the examination of the outer bends in manipulated tubing bent to small radii.

## 6. Discussion and Conclusions

Although all non-destructive testing methods discussed here have their merits and demerits, ultrasonic examination is the testing procedure, which can provide the most useful information. This can be related to the fact that all defect signals are the result of an echo from a defect which may be either of mechanical nature (cracks etc.) or of metallurgical nature (inclusions, excessive grain size). To get complete information about the quality of a tube by ultrasonic examination it is, however, necessary to have a rather sophisticated testing equipment so that defects of all types and shapes can be detected. This means that for relatively thin-walled, small size tubing, where the calibration standard notch depth constitutes a rather big portion of the tube wall, testing for both longitudinal and transverse defects should be performed, in both cases scanning in two opposite directions. By applying such a testing method it is possible to establish a high quality level at a reasonable cost with minimized risks for injurious defects to pass. This has been proved by examination of tubes giving defect signals both above and below the rejection level, e.g. for PWR steam generator tubing.

In relation to ultrasonic examination eddy current testing is still much less selective and very sensitive for other variations in a tube than defects, variations which have little or no influence on the tube quality. This means that results from eddy current testing are more difficult to interpret than from ultrasonic examination. Indications from minor dimensional variations depending on the tube processing method are examples on irrelevant information obtained. There is, therefore, a great risk that this testing method either leads to unjustified rejections of perfect tubes or that the test results must be interpreted in relation to the records from ultrasonic testing. For seamless or welded and drawn tubes the additional information obtained from eddy current testing, after testing to a strict ultrasonic specification, is minute and not worth the additional costs for testing and evaluation of results.

Finally, liquid penetrant testing being a slow, dirty and expensive testing method should be applied only when electric tests cannot give sufficient information about the tube quality. Such examples are primarily heavy wall tubing and, in the case of heat exchanger tubing, manipulated tubes, e.g. the outer surface of U-bends.

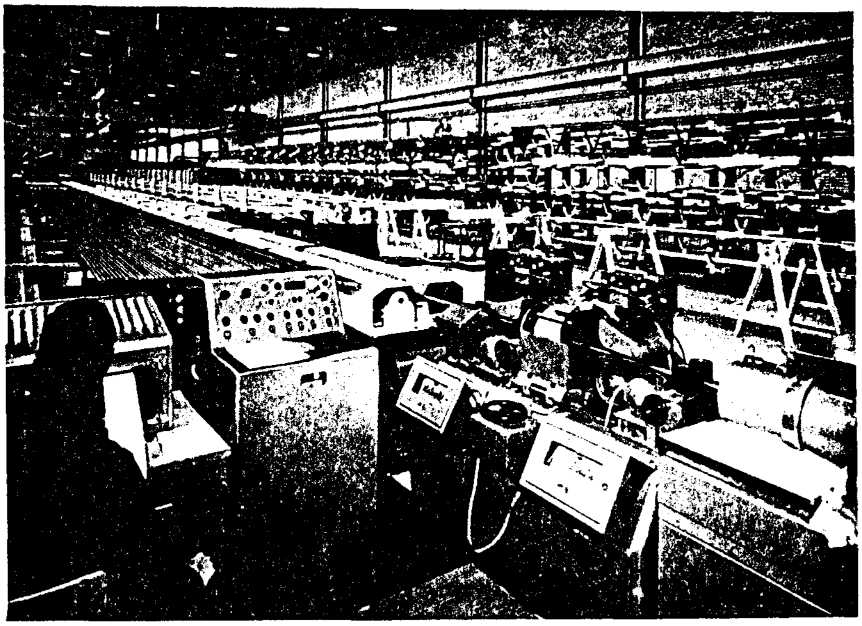

Fig. 1a

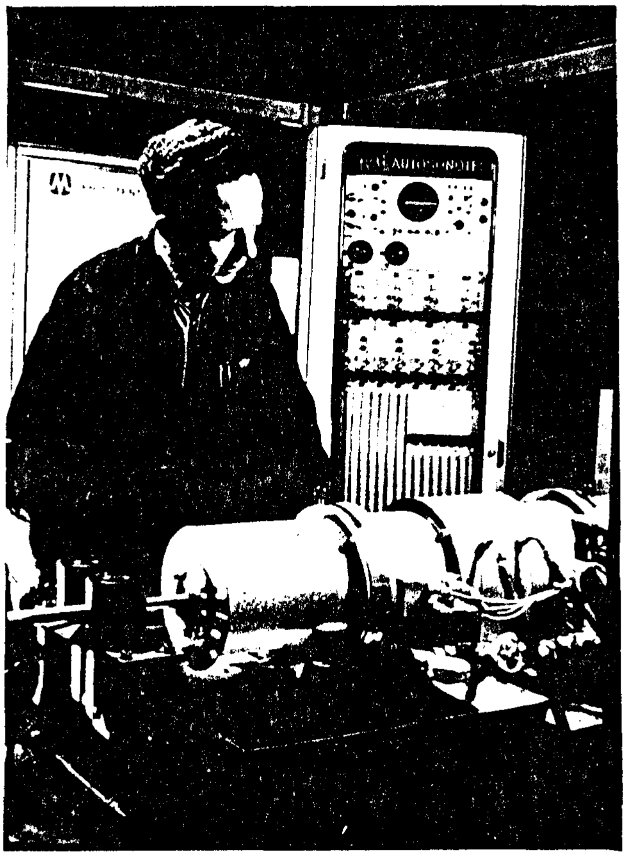

Fig. 1b

ULTRASONIC DIGITAL THICKNESS GAUGE

| | |
|---|---|
| Method: | Pulse echo with rotating transducer |
| Rotation speed: | 1500 RPM |
| Sampling rate: | 3000 times per second |
| Testing frequency: | 10 MHz |
| Minimum thickness: | 0.4 mm (0.016") |
| Accuracy: | ± 0.01 mm (± 0.0004") |
| Outputs: | High-Low alarm (adjustable set points)<br>Analog for recorder |

Fig. 2a

**Ultrasonic Digital Thickness Gauge**

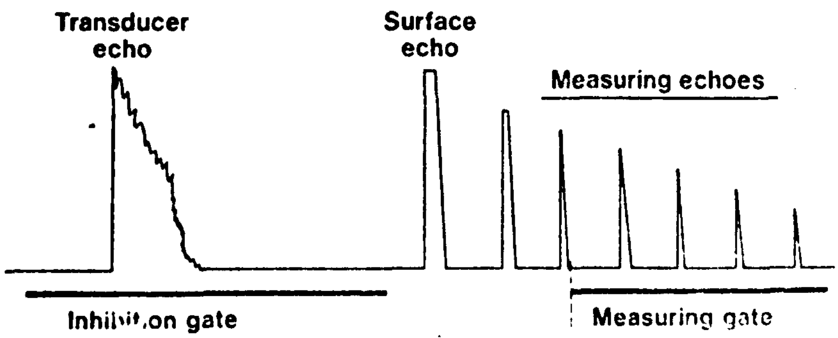


Fig. 2b